WCH-534
Rev. 0

River Corridor
Closure Contract

# Sampling and Analysis Plan for the Radiological Determination of the 618-10 Vertical Pipe Units

September 2015

For Public Release

Washington Closure Hanford

Prepared for the U.S. Department of Energy, Richland Operations Office
Office of Assistant Manager for River Corridor

Printed in the United States of America

WCH-534
Rev. 0

## CONCURRENCE PAGE

**Title:** Sampling and Analysis Plan for the Radiological Determination of the 618-10 Vertical Pipe Units

M. S. French

U.S. Department of Energy, Richland Operations Office

M S French ____________________ 8/11/15

Signature Date

B. Simes, Project Manager

U.S. Environmental Protection Agency

____________________ 8/20/15

Signature Date

DOCUMENT CONTROL 09/16/2015 ~ 8

WCH-534
Rev. 0

**STANDARD APPROVAL PAGE**

**Title:** Sampling and Analysis Plan for the Radiological Determination of the 618-10 Vertical Pipe Units

**Author Name:** W. S. Thompson

**Approval:** K. J. Koegler, VPU Remediation Project Engineer

Signature ______________________ Date 9/14/15

M. A. Buckmaster, ~~618~~-10 VPU Project Manager

Signature ______________________ Date 9/15/15

S. G. Wilkinson, Closure Operations D4/FR Project Engineering Manager

L. R. Curry for S. G. Wilkinson

Signature ______________________ Date 9/14/15

*The approval signatures on this page indicate that this document has been authorized for information release to the public through appropriate channels. No other forms or signatures are required to document this information release.*

WCH-534
Rev. 0

River Corridor
Closure Contract

# Sampling and Analysis Plan for the Radiological Determination of the 618-10 Vertical Pipe Units

September 2015


Author:

**W. S. Thompson**



Washington Closure Hanford

Prepared for the U.S. Department of Energy, Richland Operations Office
Office of Assistant Manager for River Corridor

# TABLE OF CONTENTS

**FIGURES**



**TABLES**

# ACRONYMS

| | |
|---|---|
| ALARA | as low as reasonably achievable |
| ATE | auger tool enclosure |
| CFR | *Code of Federal Regulations* |
| COC | contaminant of concern |
| COPC | contaminant of potential concern |
| CWC | Central Waste Complex |
| DOE | U.S. Department of Energy |
| DQO | data quality objective |
| EPA | U.S. Environmental Protection Agency |
| ERDF | Environmental Restoration Disposal Facility |
| ERSTI | environmental radiological survey task instruction |
| LLD | lower limit of detection |
| MDA | minimum detectable activity |
| MDL | method detection limit |
| MDP | multidetector probe |
| NDA | nondestructive assay |
| NIC | nonintrusive characterization |
| PRTR | Plutonium Recycle Test Reactor |
| QA | quality assurance |
| QC | quality control |
| QAPjP | quality assurance project plan |
| SAP | sampling and analysis plan |
| SPR | single-pass reactor |
| TMU | total measurement uncertainty |
| TRU | transuranic |
| TQ | threshold quantity |
| VPU | vertical pipe unit |
| WAC | waste acceptance criteria |
| WCH | Washington Closure Hanford |
| WIPP | Waste Isolation Pilot Plant |

# METRIC CONVERSION CHART

| Into Metric Units | | | Out of Metric Units | | |
|---|---|---|---|---|---|
| *If You Know* | *Multiply By* | *To Get* | *If You Know* | *Multiply By* | *To Get* |
| **Length** | | | **Length** | | |
| inches | 25.4 | millimeters | millimeters | 0.039 | inches |
| inches | 2.54 | centimeters | centimeters | 0.394 | inches |
| feet | 0.305 | meters | meters | 3.281 | feet |
| yards | 0.914 | meters | meters | 1.094 | yards |
| miles | 1.609 | kilometers | kilometers | 0.621 | miles |
| **Area** | | | **Area** | | |
| sq. inches | 6.452 | sq. centimeters | sq. centimeters | 0.155 | sq. inches |
| sq. feet | 0.093 | sq. meters | sq. meters | 10.76 | sq. feet |
| sq. yards | 0.836 | sq. meters | sq. meters | 1.196 | sq. yards |
| sq. miles | 2.6 | sq. kilometers | sq. kilometers | 0.4 | sq. miles |
| acres | 0.405 | hectares | hectares | 2.47 | acres |
| **Mass (weight)** | | | **Mass (weight)** | | |
| ounces | 28.35 | grams | grams | 0.035 | ounces |
| pounds | 0.454 | kilograms | kilograms | 2.205 | pounds |
| ton | 0.907 | metric ton | metric ton | 1.102 | ton |
| **Volume** | | | **Volume** | | |
| teaspoons | 5 | milliliters | milliliters | 0.033 | fluid ounces |
| tablespoons | 15 | milliliters | liters | 2.1 | pints |
| fluid ounces | 30 | milliliters | liters | 1.057 | quarts |
| cups | 0.24 | liters | liters | 0.264 | gallons |
| pints | 0.47 | liters | cubic meters | 35.315 | cubic feet |
| quarts | 0.95 | liters | cubic meters | 1.308 | cubic yards |
| gallons | 3.8 | liters | | | |
| cubic feet | 0.028 | cubic meters | | | |
| cubic yards | 0.765 | cubic meters | | | |
| **Temperature** | | | **Temperature** | | |
| Fahrenheit | subtract 32, then multiply by 5/9 | Celsius | Celsius | multiply by 9/5, then add 32 | Fahrenheit |
| **Radioactivity** | | | **Radioactivity** | | |
| picocuries | 37 | millibecquerel | millibecquerels | 0.027 | picocuries |

# 1.0 INTRODUCTION

Remediation of the 618-10 Burial Ground requires the removal, treatment, and disposal of 94 vertical pipe units (VPUs) located within the burial ground that were used for disposal of 300 Area low- to high-activity waste, including suspect transuranic (TRU) waste. Vertical pipe units that are determined to be low-level waste or mixed low-level waste will be treated, as necessary, and disposed at the Environmental Restoration Disposal Facility (ERDF). Suspect TRU or greater than Class C waste will be packaged and shipped to the Central Waste Complex (CWC) for storage or disposal.

Decisions concerning chemical contaminants, listed waste, presence of spent nuclear fuel, sodium-potassium alloys (NaK), and liquids have previously been addressed in WCH-525, *Vertical Pipe Unit Disposition for 618-10 and 618-11 Burial Grounds*. Collection of data to support the radiological determination for the VPUs (i.e., whether a VPU is low-level or suspect TRU waste) remains to be addressed and is the subject of this sampling and analysis plan (SAP).

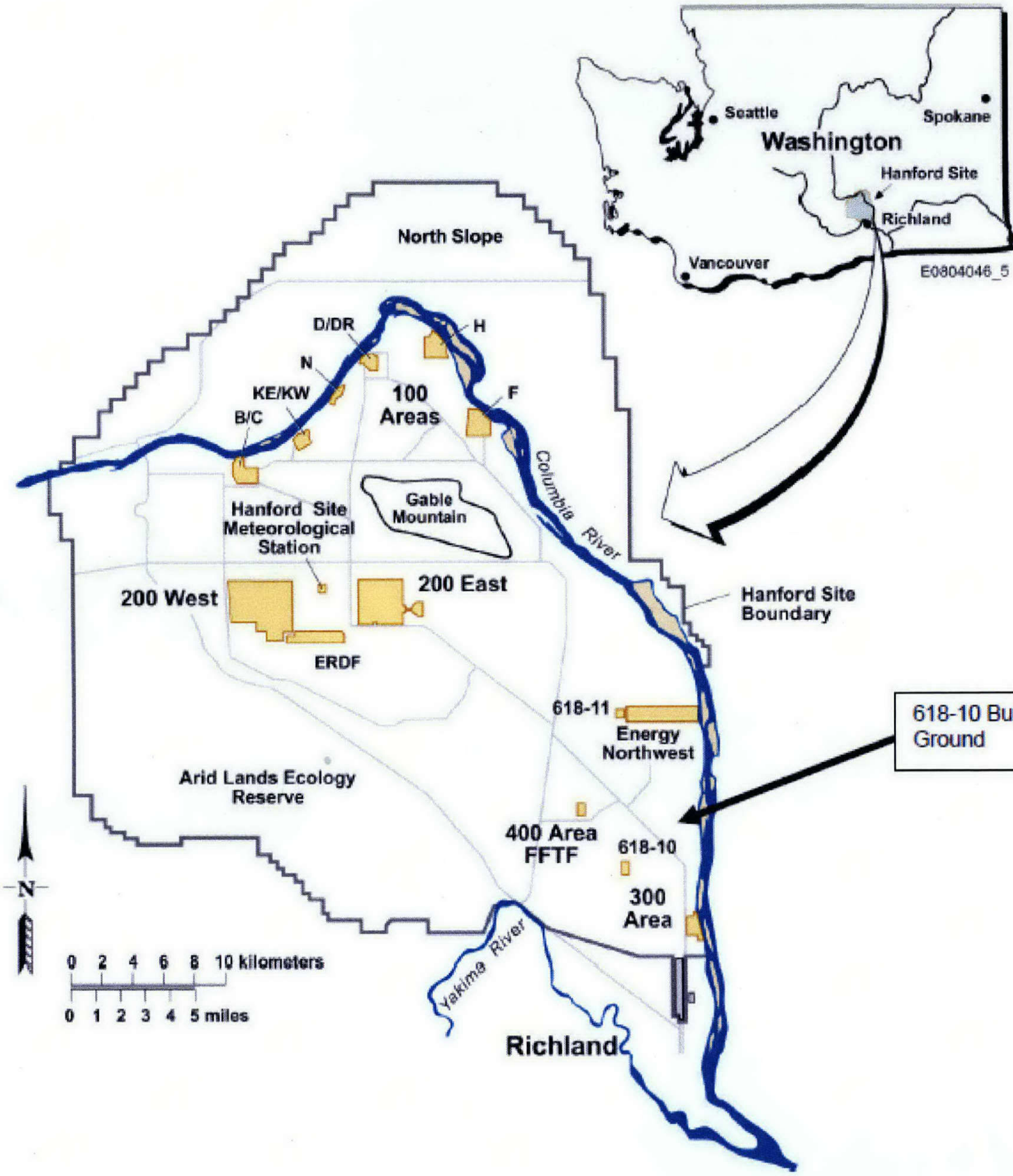
This plan provides the criteria for determining the radiological status of the VPU waste to support selection of the appropriate waste disposal path and supplements the requirements for waste characterization described in DOE/RL-2001-48, *300 Area Remedial Action Sampling and Analysis Plan*. A description of the VPUs and the proposed process for removal of the VPUs is included in Section 1.0. A summary of the data quality objectives (DQOs) that define the required decisions and the requirements for radiological determination of the VPU waste is presented in Section 2.0. The quality assurance (QA)/quality control (QC) requirements for collecting data are discussed in Section 3.0. The field characterization requirements are provided in Section 4.0.

## 1.1 BACKGROUND INFORMATION

### 1.1.1 Physical Description

The 618-10 Burial Ground (also known as the 300 Area North Burial Ground, 300 Area North, or 618-10 waste site) is located in the 600 Area of the Hanford Site in southeastern Washington State, approximately 6.9 km (4.3 mi) northwest of the 300 Area, west of Route 4 South (Figure 1). The 618-10 Burial Ground was used from March 1953 until September 1963 to dispose of low- to high-activity radioactive waste from the Hanford Site's 300 Area laboratories and fuel development facilities. The waste contained fission products and some plutonium-contaminated material.

**Figure 1. Location of the 618-10 Burial Ground.**

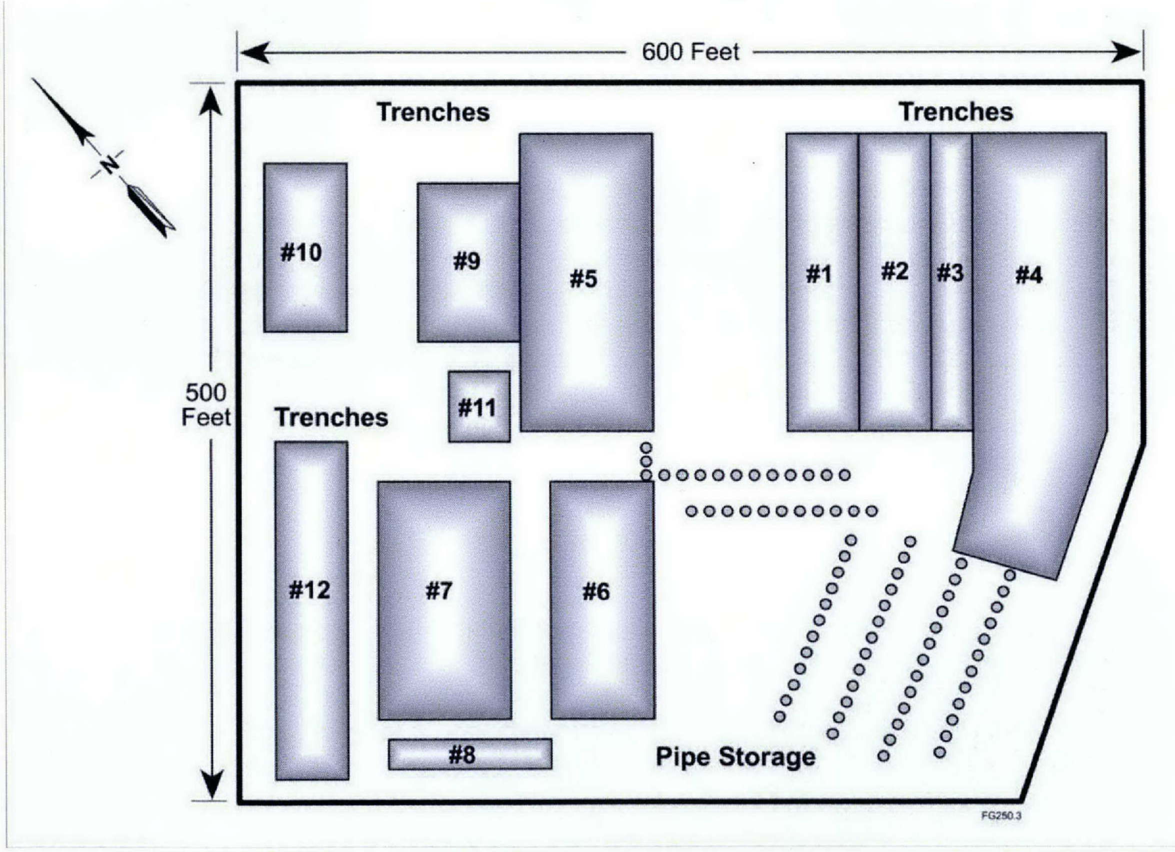


The 618-10 Burial Ground is approximately 148 by 174 m (485 by 570 ft) in size and consists of 12 trenches and 94 VPUs (Figure 2). The trenches range in size from 98 m (320 ft) long by 21 m (70 ft) wide by 7.6 m (25 ft) deep to 15 m (50 ft) long by 12 m (40 ft) wide by 7.6 m (25 ft) deep. The following three types of VPUs were used at the 618-10 waste site:

- Carbon steel pipes 25 to 61 cm (10 to 24 in.) in diameter and up to 4.6 m (15 ft) in length
- Corrugated steel pipes 36 cm (14 in.) in diameter and up to 4.6 m (15 ft) in length
- Drums, 56 cm (22 in.) in diameter and 4.4 m (14.4 ft) in length; these VPUs were constructed by welding five 208-L (55-gal) bottomless drums together end-to-end and burying them vertically.

The VPUs are generally open to the soil at the bottom and closed at the top with a concrete plug. Each VPU is covered with a minimum of 0.9 m (3.0 ft) of overburden soil.

**Figure 2. Historical Depiction of the 618-10 Burial Ground.**



### 1.1.2 Waste Disposal History

An estimated 3,670 to 5,658 $m^3$ (4,800 to 7,400 $yd^3$) of material was buried at the 618-10 Burial Ground, approximately 8.4 $m^3$ (11 $yd^3$) of which is believed equivalent to remote-handled TRU waste. Radiological and chemical hazards include cesium, strontium, plutonium, americium, neptunium, beryllium, uranium, zirconium, and sodium-potassium metals (WCH-459, *Documented Safety Analysis for Remediation of the 618-10 Burial Ground*).

The main contributors of waste to this site were 300 Area laboratory facilities. Wastes received were generated mostly by the 308 Fuels Development Laboratory, 325 Radiochemical Processing Laboratory, 321 Hydromechanical/Seismic Facility, 326 Material Science Laboratory, 327 Post Irradiation Test Laboratory, 328 Office and Maintenance Buildings, 329 Chemical Sciences Laboratory, 3211 Building, 3707 Change House, 3741 Special Machine Shop, and 3746 Irradiation Physics Building. Wastes included radioactively contaminated laboratory instruments, bottles, boxes, filters, aluminum cuttings, irradiated fuel element samples, metallurgical samples, electrical equipment, lighting fixtures, barrels, laboratory equipment and hoods, and low- and high-activity wastes sealed in containers. The exteriors of the waste containers were surveyed before the containers were transported to the 618-10 Burial Ground. The actual contents of the containers are uncertain, but radiological survey records indicate the number of waste shipments and the types of containers used. Trenches generally received low-level waste in cardboard boxes. Materials with higher radioactivity were packaged in concrete- and lead-shielded drums. Contaminated materials were often carried to the burial ground in "load luggers" (a lightly shielded transport container), which could hold between 2.8 and 5.7 $m^3$ (100 and 200 $ft^3$) of loose waste. Around 1960, the radioactivity of the waste disposed of from the 325 Radiochemical Processing Laboratory and the 327 Post Irradiation Test Laboratory hot cells increased. Cardboard containers were replaced with remote-handled "milk pails," "paint cans," and "juice cans" (Hanford Site nicknames). The containers were remotely loaded into lead-shielded casks for transport to the burial grounds. The waste was then remotely released from the casks to the VPUs. Based on review of waste disposal records, 97% of the waste disposed in the 618-10 Burial Ground VPUs came from the 327 Facility. The 327 Facility was heavily engaged in post-irradiation examination of failed single-pass reactor (SPR) fuel. Figure 3 is a schematic showing expected configuration of the typical VPU contents.

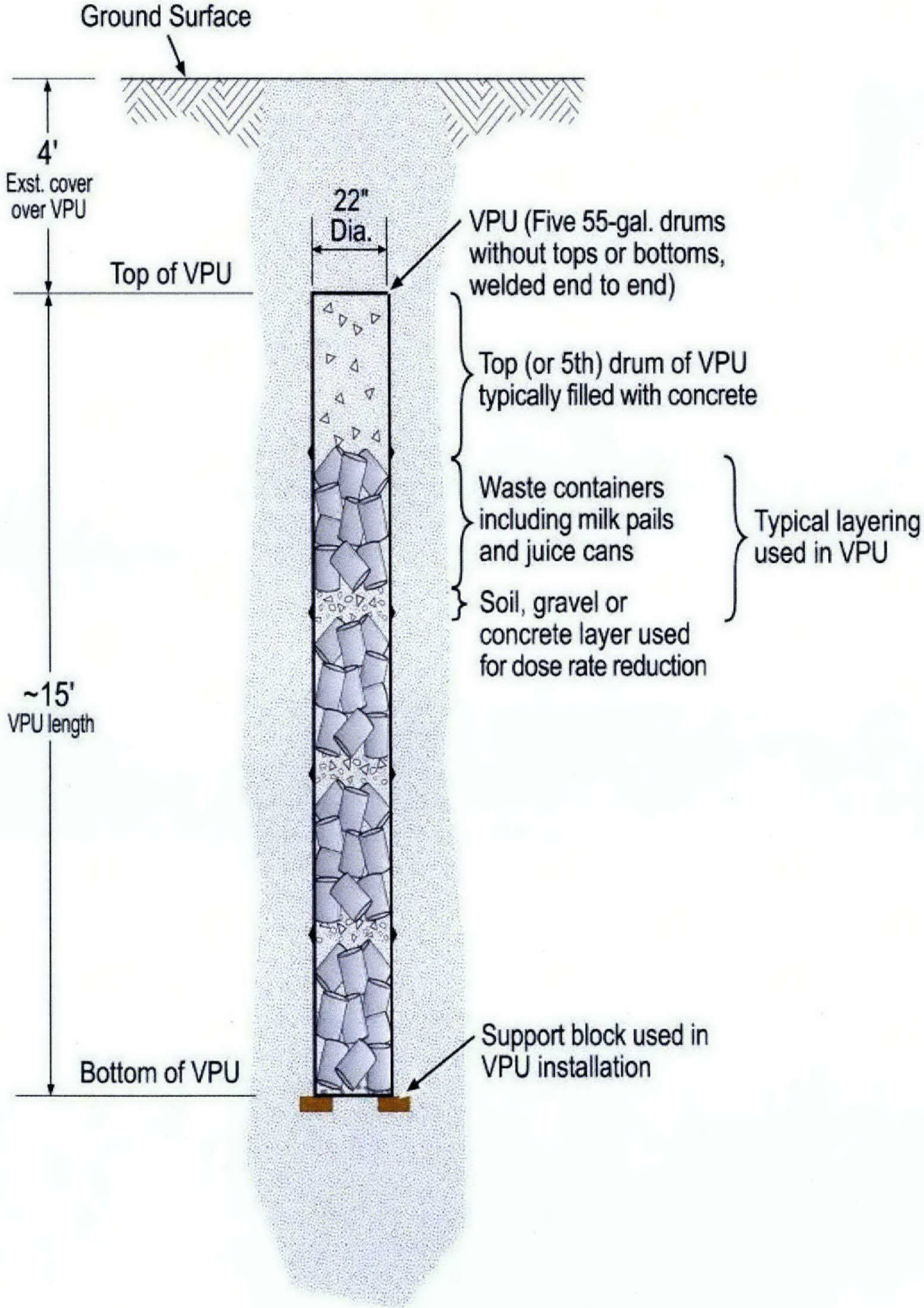
## 1.2 REMEDIATION APPROACH

The system proposed for the remediation of VPUs is described in WCH-459, *Documented Safety Analysis for Remediation of the 618-10 Burial Ground.*

The following remediation process is proposed for corrugated-pipe and drum VPUs:

- Install a steel over-casing around the VPU
- Auger to size reduce and stabilize the VPU, its contents, and mix with the soil within the over-casing that form a waste/soil matrix (hereafter referred to in this document as an augered VPU)

**Figure 3. Typical Vertical Pipe Unit Contents.**



- If the VPU is determined to be acceptable for disposal to the ERDF, grout the augered VPU
- If the augered VPU is not determined to be acceptable for disposal to the ERDF, transfer the waste into drums for further evaluation.

The steel-pipe VPUs will be remediated by an alternative method. The waste characterization and radiological determination for steel-pipe VPUs are not included in this SAP but will be addressed in a separate document.

The remediation process must meet the following requirements:

- Be compliant with applicable codes, standards, and regulations
- Meet waste acceptance criteria (WAC) for the ERDF or the CWC
- Provide for control of potential radiological particulate emissions (e.g., dust) and potential energetic releases/chemical reaction or exposure of pyrophoric wastes (e.g., sodium potassium alloy, hydrogen, zircaloy) such that any releases are within design parameters
- Maintain as low as reasonably achievable (ALARA) exposures in all operational activities, including sample collection

### 1.2.1 Over-Casing Installation

In April 2015, over-casings were driven into the ground around each of the corrugated-pipe and drum-style VPUs to a depth below the bottom of the VPU (Figure 4). The over-casing is a 1.2-m (4-ft)-diameter, 1.3-cm (0.5-in.)-thick carbon-steel pipe that is approximately 8.6 m (28.3 ft) in length. A crane with a vibratory hammer was used to drive the casing into the ground. Approximately 1.1 m (3.5 ft) of over-casing remains above the ground to provide a safety barrier during the subsequent remediation phases.

### 1.2.2 Augering of the Waste

After installation of the over-casing, the contents of the VPU will be size reduced using an auger (Figure 5). An auger tool enclosure (ATE) will be installed on top of the casing once the casing is in place. The ATE provides for dust control during augering. The auger will mix the size-reduced VPU material with the soil within the over-casing and ensure that the VPU and all containers and contents (e.g., vials, cans, bottles) inside the VPU are breached. The equipment required for this operation includes a drilling rig, drilling tools, and ATE. The drilling rig will lower and rotate the auger while traveling down through the soil overburden and into the VPU. Stabilization and size reduction of the VPU contents and mixing with the surrounding soil within the over-casing will continue as the auger traverses down the entire length of the VPU to a point below the bottom of the VPU. The entire contents of the VPU will become a size-reduced mix of original VPU items and soil within the 1.2-m (4-ft)-diameter over-casing.

**Figure 4. Vertical Pipe Unit Over-Casing Installation.**

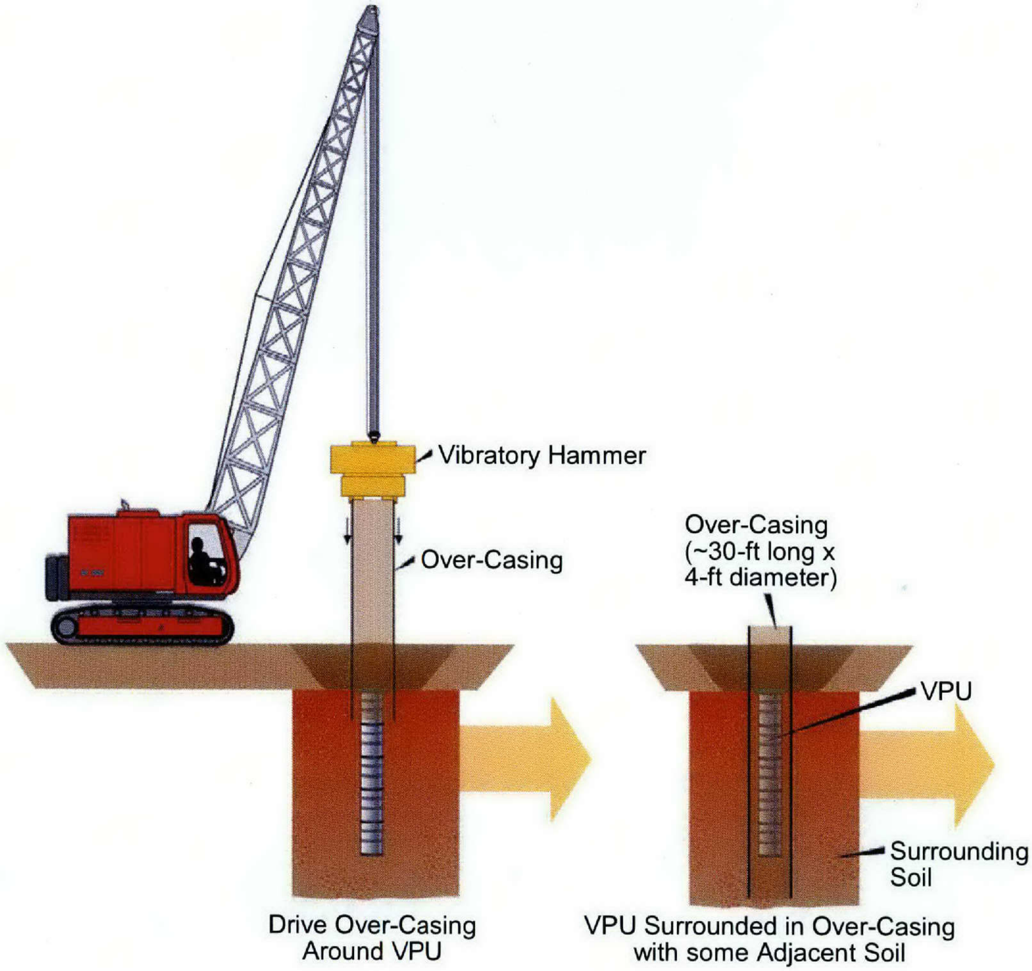
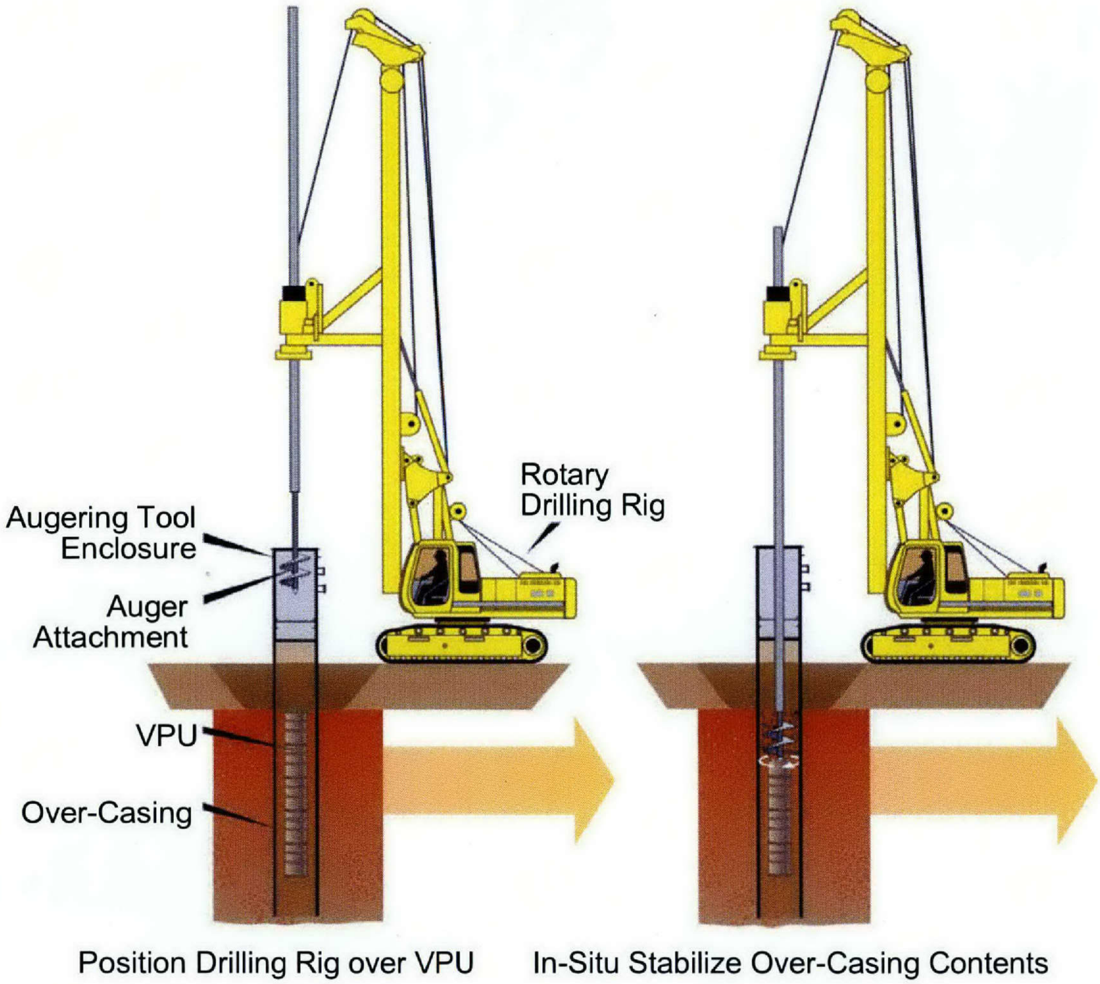


**Figure 5. Vertical Pipe Unit In Situ Stabilization and Size Reduction.**

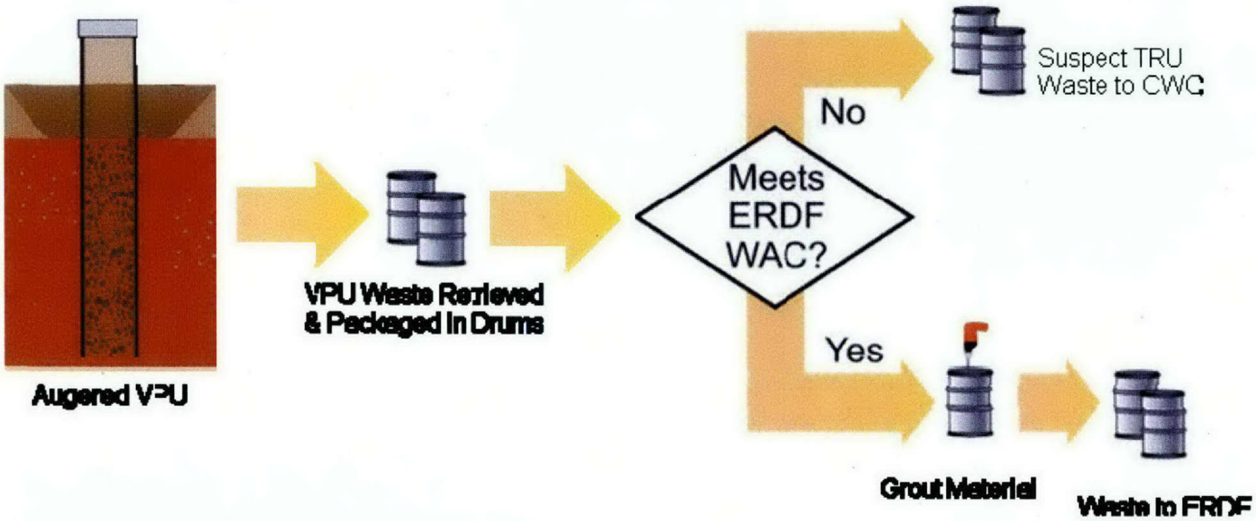



### 1.2.3 Radiological Characterization

Radiological characterization will be necessary to support a decision whether or not the augered VPU will meet the ERDF WAC (WCH-191, *Environmental Restoration Disposal Facility Waste Acceptance Criteria*). The purpose of this SAP is to provide the logic and define the characterization criteria to support this decision.

If the augered VPU is determined to meet the ERDF WAC (WCH-191) for radiological constituents, grout will be introduced and mixed with the contents. Requirements for grouting are specified in *618-10 Burial Ground VPU Treatment Plan* (WCH 2015). If it is not determined that the augered VPU meets the ERDF WAC (e.g., suspect TRU or greater than Class C waste), the augered VPU will be retrieved and transferred into drums (Figure 6). The drums filled with the augered VPU will be radiologically characterized. Drums that contain waste that is determined to meet the ERDF WAC will be grouted and shipped to the ERDF. Drums that contain waste that are not determined to meet the ERDF WAC will be shipped to the CWC.

**Figure 6. Retrieval and Drum Processing for Suspect Transuranic Waste.**




## 1.3 PROCESS KNOWLEDGE

Extensive information is available concerning the types of waste, the processes generating the waste, the containers used to store/transport waste, and the stabilization/treatment of specific wastes such as acids and reactive materials (WCH-525, *Vertical Pipe Unit Disposition for 618-10 and 618-11 Burial Grounds*). However, little is known about the exact contents of the historical waste shipments to an individual VPU. The radioactive and chemical composition of the wastes has been evaluated using historical information and assessments from past and ongoing burial ground remediation work. This reconstructed history combined with actual burial ground remediation experience was used to form a basis for process knowledge for the VPUs. A detailed discussion of the chemical and radiological process knowledge is provided in WCH-525 and was used to construct a bounding waste document for designation and profiling the 618-10 Burial Ground VPUs.

### 1.3.1 Physical Form

The VPUs are believed to be primarily filled with debris; however, soil, gravel, and concrete were added to reduce personnel exposure. The initial volume ratio of debris to soil is unknown, but could be greater than 50% soil; the final remediation processed form will be "soil-like" following the size reduction and mixing by the auger. The waste matrix will be composed of iron-based alloys, metals, lead, plastics, rubber, cellulosic material, soil, gravel, concrete, glass, ceramics, diatomaceous earth, irradiated specimens, irradiated hardware, gelatin, and treated/absorbed liquids. Liquids and/or reactive material will be incorporated into the matrix as part of stabilization during the remediation process.

### 1.3.2 Chemical Composition

The chemical composition for waste disposed to the 618-10 Burial Ground is discussed in WCH-525 and is not discussed in this SAP.

### 1.3.3 Radiological Composition

Process knowledge for the radiological composition of the 618-10 Burial Ground VPUs is provided in WCH-525, and is restated here because this knowledge provides the fundamental support for the subsequent development of the radiological DQOs (Section 2.0) and the logic for the characterization strategy (Sections 3.0 and 4.0) for the VPUs. WCH-525 also provides information supporting the determination that spent nuclear fuel is not present in the VPU waste streams.

The radionuclide inventory for the 618-10 Burial Ground VPUs was developed in the calculation titled *Radiological Inventory in the 618-10 Vertical Pipe Units* (0600X-CA-N0083). The detailed radiological inventories for the various waste categories are tabulated in *618-10 and 618-11 Burial Ground Radiological Source Terms* (0600X-CA-N0100).

As stated in WCH-525, the radioactive source term associated with the 618-10 Burial Ground was calculated by identifying the individual waste items disposed, determining the waste stream(s) within each waste item, calculating the mass of each waste stream within each waste item, calculating the isotopic activities within each waste stream item, and then summing the isotopic activities. The 0600X-CA-N0100 calculation was used to support the safety basis and involved six distinct phases:

1. Reference review
2. Fundamental data
3. Waste stream calculations
4. Dose rate to waste mass calculations
5. Waste mass to isotopic activity calculations
6. Threshold quantity (TQ) calculations.

The reference review phase included obtaining and archiving the references applicable to the 618-10 Burial Ground, extraction of key information from each reference, and entry of the information into a specifically designed Microsoft® Access® Waste Items database.

The fundamental data phase included developing the physical constraints, atomic and elemental masses, specific activities, and material densities required by the subsequent phases.

The waste stream calculations phase included developing those characteristics of the radioactive waste streams needed to input the remaining phases. Activity-per-gram calculations were performed using Oak Ridge National Laboratory ORIGEN2 software and were modeled using 105-N Reactor (N Reactor) characterization data.

The dose rate to waste mass calculations phase converted dose rate measurements associated with certain waste items into the mass of a particular waste stream responsible for the measured dose rate. The conversion process made use of the characteristics of the radioactive

® Microsoft and Access are registered trademarks of Microsoft Corporation in the United States and other countries.

waste stream within the waste item, the physical configuration of the waste item at the time of the dose rate measurement, and the matrix of the waste within the waste item. These factors were combined to generate a dose rate per gram waste versus distance curve for each unique situation. The curves were used to interpolate the dose rate from a gram of waste at the appropriate distance. The measured dose rate divided by the dose rate per gram then represents the mass of waste in the waste item. Dose rate per gram conversion factors were calculated using Los Alamos National Laboratory Monte Carlo Neutron Particle computer code. Dividing the dose rate measurements by the appropriate conversion factors was performed using a modified version of the Field Remediation Waste Items database.

The waste mass to isotope activity calculations phase converted waste mass values into isotope activities. Waste mass values were either entered during the reference review phase as waste entries, or calculated as part of the dose rate to waste mass calculations phase. The mass of each waste stream within a waste item was multiplied by a vector of isotope activities per gram from the waste stream calculation phase.

The TQ calculations phase converted a vector of isotope activities into a single TQ value that is used to determine the category of a facility. The TQ calculations were performed on both a waste item and an entire trench basis.

The total radionuclide inventory for the 618-10 Burial Ground VPUs is found in Table 16 of 0600X-CA-N0083, *Radiological Inventory in the 618-10 Vertical Pipe Units*. The inventory was calculated using a structured query language inquiry of the Waste Items database and represents the total inventory for all 618-10 VPUs. This inventory is used in the project safety analysis and provided isotopic scaling factors for use in preliminary dose modeling, waste classification, and transportation calculations. This inventory will also be used for providing isotopic scaling factors for modeling dose and calculating activities for the VPUs.

## 1.4 PREVIOUS INVESTIGATIONS

Previous investigations at the 618-10 Burial Ground included geophysical surveys, cone penetrometer investigations, and test pit characterizations. Groundwater monitoring is also routinely performed as part of the 300-FF-5 Operable Unit work scope, which includes groundwater beneath the 300-FF-2 Operable Unit.

### 1.4.1 Geophysical Surveys

Extensive geophysical surveys of the 618-10 Burial Ground have been performed, including detailed ground-penetrating radar and magnetometer surveys (BHI-00291, *Geophysical Investigations of the 618-10 and 618-11 Burial Grounds, 300-FF-2 Operable Unit*; *618-10 Burial Ground Geophysical Summary Focusing on Metal Concentrations* [WCH 2009a]). The results of these surveys were documented on site maps using a global positioning system generated coordinate system.

### 1.4.2 Nonintrusive Characterization

A direct-push drilling rig was used with a multidetector probe (MDP) assembly to provide in situ analysis of radionuclide activity within the geophysical anomalies in the burial ground and surrounding the VPUs as part of a nonintrusive characterization (NIC) activity (DOE/RL-2008-27, *Sampling and Analysis Plan for 618-10 and 618-11 Nonintrusive Sampling*).

The MDP system was configured with five standard commercial radiation detectors including a gross rate meter, two gamma spectroscopy detectors, two neutron detectors, and associated vendor software.

Four cone penetrometer tubes were installed approximately 15.2 cm (6 in.) from the outer edge and equally spaced around the perimeter of each VPU with the penetrometers extending to a depth of 0.6 m (2 ft) below the VPU. Of the five detectors present in the MDP, the majority of the data captured was gross gamma from the AMP-100 GM detector, or equivalent. Due to the lack of isotopic speciation and density compensation in the NIC data, the gross gamma data provide only an estimate of the relative dose rates. However, as part of the DQO process (Section 2.0), these data were determined to be useful for ranking the VPUs based on their average measured dose rates and are useful for guiding the selection of VPUs for further characterization.

A direct-push rig was also used to push 2-3/4-in.-outer diameter tubing to a depth approximately 0.6 m (2 ft) below the bottom of 10 of the VPUs and approximately 15.2 cm (6 in.) outside the VPU wall. After removing the inner drive tips and rods, clear PVC liners inside a sampling device were driven an additional 0.6 m (2 ft) for each of these push probes to obtain soil samples. Good soil recovery was obtained for each of these 10 samples. The samples were submitted for laboratory analysis. Field radiological surveys of these soil samples did not detect radiological contamination. A review of the laboratory analyses did not identify any chemical constituents that exceed soil cleanup criteria. Radionuclides were not detected exceeding soil cleanup criteria with the exception of a single detection of strontium-90 at 17 pCi/g in one sample and carbon-14 at 67 pCi/g in a second sample.

### 1.4.3 Intrusive Characterization

An intrusive characterization investigation of the 618-10 Burial Ground trenches was performed in August and September 2010 (WCH-431, *Field Investigation Report for the 618-10 Burial Ground Intrusive Sampling*). Five test pits were excavated across a subset of the 12 burial ground trenches located in the 618-10 Burial Ground and materials were sampled during the excavation. The purpose of the excavation was to provide data to correlate with the NIC data and the geophysical data; to demonstrate remediation and material handling methodologies; to obtain physical, chemical, and radiological information; and to provide lessons learned for future remediation work. During the excavation, several anomalies were identified, including depleted uranium drums, oil-filled drums, concreted drums, laboratory bottles and debris, metal debris, a shielded cask, and metal pipes. Soil samples were collected from within each test pit. Nondestructive assay (NDA) was performed on several concreted drums. The results of this investigation are provided in WCH-431.

### 1.4.4 Groundwater Monitoring

Groundwater contamination associated with the 618-10 Burial Ground is addressed within the 300-FF-5 Operable Unit scope. There currently is no groundwater contamination attributed to the 618-10 Burial Ground.

# 2.0 DATA QUALITY OBJECTIVES

This section provides a summary of the DQOs. The DQO process is a quality management tool developed by the U.S. Environmental Protection Agency (EPA) that is used to facilitate the planning of data collection activities (EPA/240/B-06/001, *Guidance on Systematic Planning Using the Data Quality Objectives Process* [EPA QA/G-4]). The DQO process was used to develop clear and concise study objectives, define the appropriate type of data required, and specify tolerable levels of potential decision errors that will be used as the basis for establishing the quality and quantity of data needed to support decisions.

Data quality objective scoping interviews were held with Washington Closure Hanford (WCH), U.S. Department of Energy (DOE), and EPA personnel the week of March 1, 2011 and October 4, 2011, with a DQO workshop held on January 12, 2012. The purpose of the workshop was to finalize the radiological characterization process for determining whether the VPUs at the 618-10 Burial Ground meet low-level radioactive waste criteria and can be disposed at the ERDF. A list of DQO participants is provided in Table 1.

**Table 1. 618-10 Burial Ground Data Quality Objective Workshop Participants. (2 Pages)**

| Name | Organization | Role | Interview/Meeting Dates | | |
|---|---|---|---|---|---|
| | | | Week of March 1, 2011 | Week of October 4, 2011 | January 12, 2012 |
| Warren Bryant | WCH | Project Manager | x | x | x |
| Mike Casbon | WCH | ERDF Resident Engineer | x | -- | x |
| Mike Collins | DOE | Waste Management | x | -- | x |
| Dave Einan | EPA | ERDF Project Manager | x | x | |
| Larry Gadbois | EPA | 300 Area Project Manager | x | x | x |
| Darrin Faulk | WCH | Environmental Lead | x | x | x |
| Dennis Faulk | EPA | Project Manager | x | x | x |
| Bob Hynes | WCH | Waste Management/ Transportation | x | x | x |
| Dan Haggard | WCH | Nondestructive Analysis | -- | x | x |
| Kim Koegler | WCH | Project Engineer | x | x | x |
| Catherine Louie | DOE | Project Manager | -- | x | x |
| Paul Macbeth | DOE | Nuclear Safety & Transportation | -- | -- | x |
| Dale Obenauer | WCH | Project Engineer | x | x | x |
| Chuck Ramsey | EnviroStat | DQO Facilitator | x | x | x |
| Wendy Thompson | WCH | DQO/Sampling | x | x | x |
| Steve Wilkinson | WCH | Field Remediation Project Engineer | x | x | x |

**Table 1. 618-10 Burial Ground Data Quality Objective Workshop Participants. (2 Pages)**

| Name | Organization | Role | Interview/Meeting Dates | | |
|---|---|---|---|---|---|
| | | | Week of March 1, 2011 | Week of October 4, 2011 | January 12, 2012 |
| Rich Weiss | WCH | Analytical Laboratories | x | x | -- |
| Ames Zacharias | WCH | Radiological Engineer | x | x | x |
| Jamie Zeisloft | DOE | 618-10 Project Manager | x | x | x |

DOE = U.S. Department of Energy
DQO = data quality objective
ERDF = Environmental Restoration Disposal Facility
EPA = U.S. Environmental Protection Agency
WCH = Washington Closure Hanford
X = indicates interview or meeting attended

## 2.1 DATA QUALITY OBJECTIVE STEP 1: STATE THE PROBLEM

### 2.1.1 Problem Statement

Historical information indicates that the VPUs contain radioactive and chemically contaminated waste materials, including possible TRU material. This waste will be in the form of debris, soil, concrete, and may include containers with liquid contents. The material in the VPUs requires radiological characterization to determine the proper disposal facility.

## 2.2 DATA QUALITY OBJECTIVE STEP 2: IDENTIFY THE DECISION

Decision Statement:

Determine if the augered VPU meets the radiological WAC for disposal to the ERDF. The radionuclide criteria in the ERDF WAC exclude waste that is high level, transuranic, greater than Class C waste, spent nuclear fuel, by-product material, special nuclear material, and fissile material as defined in Section 1.3 of the ERDF WAC (WCH-191).

Alternative Actions:

- If the augered VPU does not meet the radiological WAC for the ERDF, then the augered VPU material will be removed and packaged into 208-L (55-gal) drums. The drums generated by this process will be further evaluated using NDA to determine if the individual drum meets the ERDF WAC or if disposal must occur in a facility other than the ERDF. Additionally, this material must be retrieved from the VPU and packaged to meet the alternative disposal facility WAC. This alternative disposal is transfer of the waste to the CWC, with eventual disposal to the Waste Isolation Pilot Plant (WIPP).

- If the augered VPU meets the ERDF radiological WAC, the material can be grouted and disposed in the ERDF.

## 2.3 DATA QUALITY OBJECTIVE STEP 3: IDENTIFY THE INPUTS TO THE DECISION

The data inputs needed to resolve the decision statement (Section 2.2) were identified along with areas where additional data collection is required. The DQO scoping meetings were used to discuss and evaluate the usability of the existing information and to develop logic supporting the selection of additional data requirements and collection and measurement methods. The consensus of the DQO team was that additional information concerning the radiological content for an augered VPU is needed in order to determine the proper disposal path.

### 2.3.1 Identify the Information Required

In order to determine if the augered VPU meets the radiological WAC for disposal to the ERDF, information concerning the radiological isotopes present in the waste and the activity of each isotope is needed. In addition, the action level, or threshold value, that provides the criterion for choosing between the alternative actions is needed. Since this SAP involves only the radiological determination for the VPUs, only the action levels associated with deciding whether the waste is low-level radioactive for acceptance at the ERDF (WCH-191) or TRU/greater than Class C for acceptance at the CWC (HNF-EP-0063, *Hanford Site Solid Waste Acceptance Criteria*) are needed.

**2.3.1.1 Action Levels.** Waste is certified for disposal when it can be determined that the WAC for the disposal facility have been satisfied. Since there are two possible waste disposal facilities (ERDF or CWC/WIPP), the action levels for each must be identified:

- ERDF Action Level: For waste to be disposed at the ERDF, the augered VPU must meet the ERDF radiological WAC including having a TRU activity of 100 nCi/g or less and be considered not greater than U.S. Nuclear Regulatory Commission Class C waste as defined in 10 *Code of Federal Regulations* (CFR) 61.55, "Waste Classification."

- CWC Action Level: Waste that is not acceptable for disposal at the ERDF will be transferred to the CWC for storage until eventual disposal at WIPP. The action level for acceptance at the CWC is TRU activity greater than 100 nCi/g.

**2.3.1.2 Contaminants of Potential Concern/Contaminants of Concern.** The contaminants of potential concern (COPCs) for the 618-10 Burial Ground are identified in DOE/RL-2001-48, *300 Area Remedial Action Sampling and Analysis Plan*. The process for identifying COPCs for the burial ground trenches and VPUs relied on a search of historical documents and IHE-2009-0006, *Integrated Chemical and Radiological Hazard Evaluation Worksheet – 618-10 Burial Ground/North Burial Ground* (WCH 2009b). The initial COPC list was then screened using exclusion criteria to develop the contaminant of concern (COC) list (DOE/RL-2001-48). Radionuclides with half-lives less than 3 years (and no significant "daughters"), naturally occurring radionuclides associated with background radioactivity (e.g., potassium-40, thorium-240, thorium-232, and radium-226), and a limited number of radionuclides having no analytical method, requiring significant analytical resources, and/or can be estimated from other reported radionuclides were excluded as COCs. Table 2 provides a list of the radionuclide COCs for the 618-10 Burial Ground. Chemical COCs are not addressed in this DQO for radiological determination of the VPUs.

**Table 2. Radionuclide Contaminants of Concern for the 618-10 Vertical Pipe Units.**

| Radionuclide | Half-Life (years) |
|---|---|
| Am-241 | 432.2 |
| Cm-244 | 18.11 |
| Co-60 | 5.271 |
| Cs-137 | 30.17 |
| Eu-152 | 13.6 |
| Eu-154 | 8.8 |
| Eu-155 | 4.96 |
| H-3 | 12.33 |
| I-129 [a] | 15,700,000 |
| Np-237 | 2,140,000 |
| Pu-238 | 87.75 |
| Pu-239 | 24,131 |
| Pu-240 [a] | 6,537 |
| Pu-241 | 14.4 |
| Sr-90 [b] | 28.6 |
| Tc-99 | 213,000 |
| U-233 | 159,200 |
| U-234 [a] | 244,500 |
| U-235 | 703,800,000 |
| U-238 | 4,468,000,000 |

[a] Generate estimated activities based on calculations from the detected isotopes based on reactor fission/activation yields.
[b] Analyzed as total radioactive strontium.

**2.3.1.3 Radiological Characterization Methods.** A combination of process knowledge, calculations, sampling, NDA, in situ radiological measurements, and destructive assay (laboratory sample analysis) are available for radiological characterization of the VPUs. A balanced approach using these methods was considered as part of the DQO process to determine how additional information is to be obtained in order to make a radiological determination for the VPUs. Existing sources of information for the 618-10 Burial Ground VPUs were evaluated, including looking at process knowledge, reviewing the waste inventory (0600X-CA-N0083, 0600X-CA-N0100), performing calculations (ORIGEN2 and TQ), and using the in situ cone penetrometer data.

Available characterization methods and sources are discussed below.

- Process Knowledge and Calculations: Process knowledge includes the use of reactor fuel burn-up calculations (ORIGEN2 codes) and inspection of waste disposal records used to develop the inventory for the VPUs (0600X-CA-N0083, 0600X-CA-N0100). A review of disposal records provided information including a description of the waste, the process from which the waste was generated, dose rates, and estimates of primary radionuclides present. Additional discussion is provided in Section 1.3.3 of this SAP.

Reactor fuel burn-up calculations have been used to generate scaling factors, which will then be used to calculate the activity of the radionuclides of concern based on NDA and/or in situ measurements of the cesium-137 activity. The key parameter controlling applicability of the scaling factors is the concentration of plutonium-239 relative to the concentration of cesium-137. This ratio is a weak function of reactor design and operating conditions, but increases steadily with reactor fuel burn-up.

The waste in the 618-10 Burial Ground VPUs was predominantly generated during examination of failed SPR fuel and includes the residues from these examinations. Very little SPR fuel still existed when Hanford made the transition from weapons production to waste management, and as a result detailed composition information for SPR fuel was never developed. However, large quantities of N Reactor fuel remained to be managed at that time, so detailed composition information was developed and is now readily available. Since the plutonium-239/ cesium-137 concentration ratio is only weakly dependent on reactor design and operating conditions, N Reactor fuel data are an appropriate surrogate for SPR fuel data for the purpose of constructing scaling factors.

Based on review of waste disposal records, 97% of the waste disposed in the 618-10 Burial Ground VPUs came from the 327 Facility. The 327 Facility was heavily engaged in post-irradiation examination of failed SPR fuel. Failed fuel was immediately removed once detected to minimize contamination of the Columbia River. Therefore, failed fuel typically has much less burn-up than fuel irradiated full term. To ensure that the results were bounding, N Reactor fuel irradiated to 12% plutonium-240 was selected to construct the scaling factors, while most of the SPR fuel residue in the 618-10 Burial Ground VPUs has much less than 6% plutonium-240.

There is a small quantity of specialty fuel residue in the 618-10 Burial Ground VPUs, most notably about 15 g of plutonium from examination of Plutonium Recycle Test Reactor (PRTR) fuel. Although PRTR fuel has a higher Pu-239 to Cs-137 ratio than either SPR or N Reactor fuel, the use of scaling factors from 12% Pu-240 N Reactor fuel provides more than adequate conservatism.

- Nondestructive Assay: Nondestructive assay measures penetrating radiation emitted from containerized radioactive material. The detected radiation is related to the radionuclides present and their quantities. It is convenient, rapid, and in many cases can provide an accurate measure of radioactivity packaged in containers from 3.8 to 11 L (1 to 3 gal) in size up to 114 to 208 L (30 to 55 gal) in size, including B-25 box size containers. Nondestructive assay is widely used because of its appeal in reducing sample collection of hazardous materials. Because NDA is a nonintrusive measurement, it eliminates the need for chemical separation of isotopes from one another, material processing to reduce radiation dose rates to levels manageable by the laboratory, and management of hazardous/radioactive waste generated by destructive analysis. As a result, exposure of personnel to radiation and hazardous substances is greatly reduced. For applications in which NDA is applicable, the sampling error that otherwise is associated with sampling heterogeneous materials is negligible. Because bulk measurement by NDA describes average radioactivity of the entire container, multiple NDA measurements of the same container are not required for improving the quality of the radioactivity measurement. However, there are restrictions in using NDA; it can suffer significantly from matrix effects in large containers. Under some field conditions, NDA cannot measure the isotopes present. Therefore, NDA results are often used in combination with destructive analysis and process knowledge. This allows scaling unmeasured isotopes to measured isotopes, to make better corrections for matrix and source effects, and to adjust parameters of the measurement system to achieve an optimized response. Nondestructive assay is a three-step process: measure radiation, associate radiation with a specific radionuclide (or radionuclides), and determine the amount of each radionuclide.

- In Situ Radiological Characterization Using the Dose-to-Curie Method: The feasibility of in situ radiological characterization relies on the radionuclide-specific scaling factors developed from the process knowledge to convert the measured dose rates into radionuclide-specific activities. The quantity of plutonium in each VPU is strongly correlated to the quantity of cesium-137, and a field exposure rate instrument can be used to characterize the VPUs.

- Sampling and Laboratory Analysis (Destructive Assay): Collecting samples for laboratory analysis can be used to provide an entire radiological profile of a waste stream, including relative isotopic abundance. However, a critical limitation of laboratory analysis is the adequacy of collecting a representative sample, particularly of a heterogeneous material. Sampling heterogeneous material will likely not result in a high level of representativeness; therefore, implementing highly rigorous QC acceptance criteria on the laboratory radiochemical analysis adds little value to the actual quality of the data. Furthermore, the risk and cost of collecting samples of highly radioactive material for laboratory analysis should be factored into the sampling strategy. Obtaining representative samples of heterogeneous material is impossible without shredding and homogenizing to the extent practical prior to sampling. Many times this size reduction cannot be performed to provide a sample size that meets the minimal size (grams) that can be analyzed using laboratory instrumentation.

## 2.4 DATA QUALITY OBJECTIVE STEP 4: DEFINE THE STUDY BOUNDARIES

There are two populations of interest for the 618-10 Burial Ground, an augered VPU and a 114-L (30-gal) or 208-L (55-gal) drum. All corrugated-pipe and drum VPUs are considered part of the study, but the two populations of interest are (1) each augered VPU, and (2) a waste drum, as generated.

## 2.5 DATA QUALITY OBJECTIVE STEP 5: DECISION RULES

The following decision rules were developed to support the actions taken concerning radiological disposal pathways for an augered VPU and a waste drum:

- If the augered VPU is determined to be low-level waste (per the ERDF WAC), then the augered VPU will be grouted and shipped to the ERDF; otherwise, the augered VPU will be retrieved and transferred into drums for further evaluation.

- If the retrieved drum is determined to be low-level waste (per the ERDF WAC), then the material will be grouted and shipped to the ERDF; otherwise, the drum will be shipped to the CWC.

## 2.6 DATA QUALITY OBJECTIVE STEP 6: SPECIFY TOLERANCE LIMITS ON DECISION ERRORS

The most severe decision error for characterizing the VPUs is shipping material to the ERDF that does not meet the ERDF WAC. This error is controlled by the following:

- Using the best practicable method to characterize a VPU; this was determined to be use of a combination of process knowledge, calculations, and an in situ radiological measurement method.

- After the low-level waste determination is made, grout will be added for treatment of lead. Adding grout for treatment of lead during the remediation process increases the confidence of the low-level waste decision, since the grout is not part of the mass for the initial waste classification of the VPU.

## 2.7 DATA QUALITY OBJECTIVE STEP 7: SAMPLE DESIGN OPTIMIZATION

The information from the previous six DQO steps was reviewed to develop an optimal design for obtaining information needed for radiological characterization of the VPUs. An optimal design is one that satisfies the DQOs, provides maximum achievable ALARA compliance, and collects the information using the most cost-effective approach.

During step 3 of the DQO process, it was decided that additional data were needed to supplement existing process knowledge, calculations, and proposed radionuclide inventory.

The following was agreed to:

- Additional data should be collected.

- An augered VPU is laterally uniform, but vertically stratified.

- Nondestructive assay was selected over laboratory analysis due to homogeneity concerns and insufficient sample size associated with the traditional laboratory analysis.

- The NIC data are useful for ranking the VPUs based on their average measured dose rates.

As planning for remediation of the VPUs proceeded, and concerns for worker exposure and contamination control associated with sampling the VPUs were addressed, EPA and DOE recognized that alternative methods for radiological characterization should be evaluated. In situ radiological characterization was identified as a method that could be used for the VPUs and meet the DQOs resulting from the January 12, 2012 DQO workshop (WCH 2014).

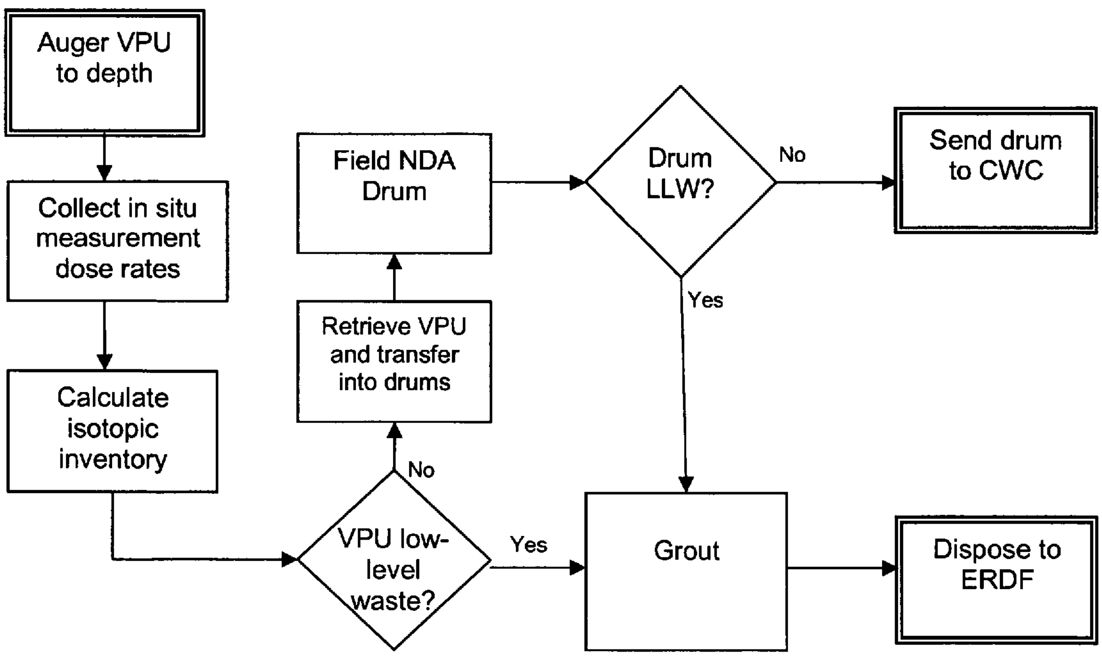
Characterization of a limited number of VPUs with a well-defined and controlled process of in situ radiological measurements can provide a cost-effective and ALARA methodology for radiological characterization of the VPUs. Equally, in situ radiological characterization has been determined to be a viable method to meet the DQOs (WCH 2014). Isotopic distributions determined from process knowledge and the radionuclide inventory can be used to develop scaling factors to associate hard-to-measure radionuclides to in situ radiological measurements.

### 2.7.1 In Situ Radiological Characterization

Data derived from the in situ radiological characterization of a VPU will be used to build the radionuclide waste inventory for the VPU and determine if the VPU waste meets the ERDF WAC. Because the dose to curie method does not supply radionuclide-specific information, scaling factors for the radionuclide ratios are developed from the process knowledge described in Section 1.3.3 and used to convert measured dose rate into radionuclide-specific activities.

Augered VPUs that are determined to be suspect TRU or greater than Class C will be retrieved and transferred into drums and a field NDA will be used to evaluate each drum as currently used for characterization of drums removed from the 618-10 trenches (WCH-449, *618-10 Burial Ground Drum Sampling and Analysis Instruction)*. Drums that are determined to be suspect TRU waste will be shipped to the CWC facility. The decision logic for VPUs is presented in Figure 7.

**Figure 7. Decision Logic for Vertical Pipe Units.**



### 2.7.2 Selection of Vertical Pipe Units for Radiological Characterization

The selection of VPUs for in situ radiological characterization will be based on using the NIC data (DOE/RL-2008-27, North Wind 2010) for ranking the VPUs as specified in calculation 0600X-CA-N0098, *618-10 Vertical Pipe Units Ranked by TRU Concentration.* The process is as follows:

- Rank the VPUs using an analysis of the NIC data and the inventories developed for the 618-10 Burial Ground.
- Based on the ranking, select two consecutively ranked VPUs to characterize using the in situ radiological measurement process.
- Evaluate the results and determine if each VPU meets the ERDF WAC (pass) or does not (fail). The possibilities for the two consecutively ranked VPUs are pass/pass, pass/fail, fail/pass, or fail/fail.
- At the ranking point where two consecutively ranked VPUs pass, then subsequent lower ranked VPUs are thereby determined to all pass and do not require in situ radiological characterization.
- If at least one of the two consecutively ranked VPUs fails, then VPUs above that ranking point would be considered to fail unless specifically characterized.
- A minimum of four VPUs will be characterized.

# 3.0 QUALITY ASSURANCE PROJECT PLAN

This quality assurance project plan (QAPjP) identifies the policies, organizations, objectives, activities, methods, and QA/QC procedures for collecting data to support waste characterization as described in Section 2.0.

This QAPjP follows the EPA guidelines contained in EPA/240/R-02/009, *Guidance for Quality Assurance Project Plans* (EPA QA/G-5), and EPA/240/B-01/003, *EPA Requirements for Quality Assurance Project Plans* (EPA QA/R-5).

Characterization activities will be performed in accordance with requirements cited in the following documents:

- 10 CFR 830, "Nuclear Safety Management," Subpart A, "Quality Assurance Requirements"
- DOE O 414.1D, *Quality Assurance*
- EPA/240/B-01/003, *EPA Requirements for Quality Assurance Project Plans* (EPA QA/R-5)
- DOE/RL-96-68, *Hanford Analytical Services Quality Assurance Requirements Documents.*

## 3.1 PROJECT DESCRIPTION

The characterization strategy described in this SAP (Section 2.0) will be used to provide data to support waste classification for the disposal of the 618-10 Burial Ground VPUs.

## 3.2 PROJECT ORGANIZATION AND RESPONSIBILITIES

Washington Closure Hanford (WCH) has overall responsibility for the in situ radiological characterization effort. The project organization is depicted in Figure 8.

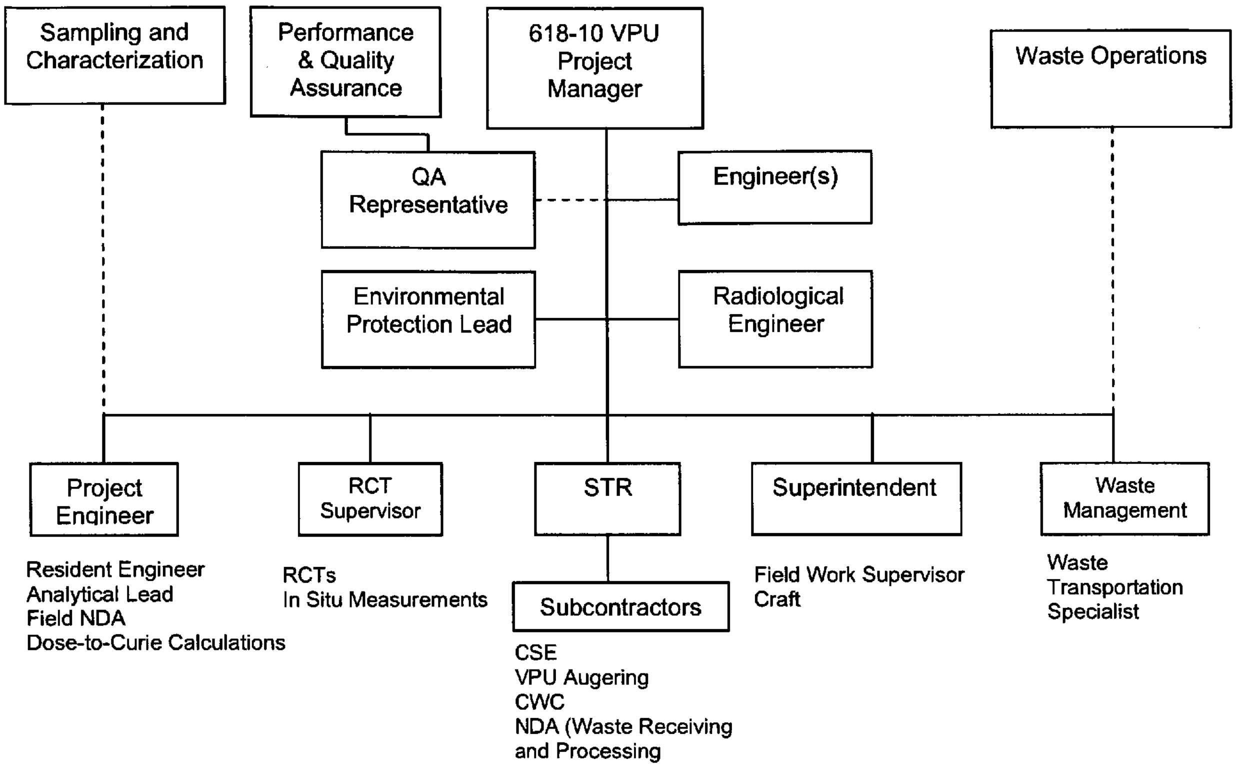
### 3.2.1 WCH Project Management

The WCH Closure Operations D4/FR Project will provide project management, project engineering, and coordination of field support functions to support implementation of this SAP.

The WCH project team will:

- Provide project, task, and engineering management necessary to carry out tasks
- Act as a liaison to contractor functional organizations, as required
- Prepare work packages to support characterization
- Conduct and document pre-job meetings supporting in situ radiological characterization
- Provide field support for in situ radiological characterization
- Provide field NDA services
- Provide industrial hygiene, radiological control, and safety support and monitoring for field activities including in situ radiological characterization
- Provide waste management support
- Provide subcontractor oversight.

**Figure 8. Project Organization.**




### 3.2.2 Subcontractors

A subcontractor may be used to provide services for augering the VPU and supporting the in situ VPU measurement process.

### 3.2.3 CH2MHill Plateau Remediation Company

The CH2M Hill Plateau Remediation Company is responsible for accepting TRU or greater than Class C wastes for storage/disposal at the CWC.

### 3.2.4 Sampling and Characterization Organization

The WCH Sampling and Characterization organization will provide personnel to support field activities including in situ radiological characterization, as needed.

### 3.2.5 Quality Assurance

The WCH Quality Assurance organization is responsible for performing independent QA assessments, as appropriate.

### 3.2.6 Data Users

Data collected from this characterization effort may be used by any of the following organizations:

- Closure Operations D4/FR Project
- Waste Operations
- Engineering Services
- Radiological Control
- Safety and Health
- Quality Assurance
- EPA and DOE.

## 3.3 TRAINING REQUIREMENTS/CERTIFICATIONS

Training or certification requirements needed by WCH personnel are described in BSC-1, *Business Services and Communications*, Procedure BSC-1-2.4, "Training Requirements," and WCH-314, *Sampling and Characterization Quality Assurance Program Plan*, Vols. 1 and 3. The WCH training program provides workers with the knowledge and skills necessary to execute assigned duties in a safe manner. A graded approach is used to provide DOE order and regulatory compliant training to all workers. Specialized employee training includes pre-job training, emergency preparedness, plan-of-the-day meetings, and facility/work site orientations.

Only appropriately trained and qualified personnel shall be allowed to collect, review, verify, and validate in situ radiological measurements and NDA measurements. Training requirements for operation of the instrumentation shall be based on existing industry standardized training requirements (e.g., ASTM C1490, *Standard Guide for Selection, Training, and Qualification of Nondestructive Assay [NDA] Personnel*) and shall meet the criteria identified in the following documents:

- WCH Radiological Control Technician OJT/OJE Instruction Guide, PS-OJT-52, Operation of the ISOCS Spectroscopy System

- Radiological Control training course 105595, Qualification Standard for In-Situ Object Counting System (ISOCS) Review/Analyst

- Training requirements for operation of the AMP-100 or equivalent dose rate instrument that will be used for collection of in situ radiological measurements will be specified in the instrument procedure.

Personnel performing dose-to-curie radionuclide inventory calculations will be trained in the use of the computer software and the principles of calculating radioactive source terms. This training will ensure consistency between calculations and results.

## 3.4 DATA QUALITY

The QA objective of this plan is to provide data of known and appropriate quality for the needs identified through the DQO process. Data quality is determined by assessing precision, accuracy, representativeness, comparability, and completeness (i.e., PARCC parameters). Definitions of these terms, applicable procedures, and level of effort are described below:

- Precision is a measure of the data spread when more than one measurement has been taken on the same material. Precision can be expressed as the relative percent difference for duplicate measurements.

- Accuracy is an assessment of the closeness of the measured value to the true value. For some radionuclide measurements, method calibrations against known standards are used to establish accuracy.

- Representativeness is a measure of how closely the results reflect the actual activity of radionuclides in the material being measured. Documentation will be established to show that protocols have been followed and the integrity of the dose rate measurements is ensured. The project will assess potential cross contamination from reuse of equipment (auger flights and radiological detector) between VPUs.

- Comparability expresses the confidence with which one data set can be compared to another. Data comparability will be maintained using defined procedures and consistent methods and units.

- Completeness is a measure of the amount of valid data obtained from the measurement system and the complete implementation of defined field procedures. Completeness is assessed during the data validation process.

## 3.5 FIELD DOCUMENTATION

Field documentation related to the radiological characterization is maintained in accordance with ENV-1, *Environmental Monitoring & Management,* ENV-1-2.5, "Field Logbooks."

Field documentation associated with in situ radiological measurements and onsite NDA are maintained in accordance with the following:

- WCH-314, *Sampling and Characterization Quality Assurance Program Plan, Volume 1: Administrative Requirements*

- WCH-314, *Sampling and Characterization Quality Assurance Program Plan, Volume 3: Field Analytical Technical Requirements*

- RC-300, *Radiological Control Instrumentation Procedures*, RC-300-4.3, "In-Situ Object Counting System (ISOCS) Quality Assurance"

- A procedure that will be prepared for operation of the AMP-100 or equivalent detector for collection of in situ radiological (i.e., dose rate) measurements.

## 3.6 CHANGE CONTROL

To ensure efficient and timely completion of tasks, minor changes can be made to the original work scope (outlined in this SAP) in the field by the resident engineer (or designee), provided the changes do not impact the technical adequacy of the job. Such changes will be documented with justification in a field logbook.

## 3.7 MEASUREMENT/DATA ACQUISITION

There are three phases of data collection and assessment for the radiological determination for the VPUs that are subject to QA project requirements: in situ dose rate measurements; dose-to-curie calculation; and scaling factor determinations. The detection level for in situ gamma measurements shall be such that the dose-to-curie calculation is viable at 100 nCi/g. For suspect TRU waste that is drummed, field NDA is also subject to QA requirements.

The following subsections present quality objectives for characterization data. The requirements for instrument calibration, maintenance supply inspections, and data management are also discussed.

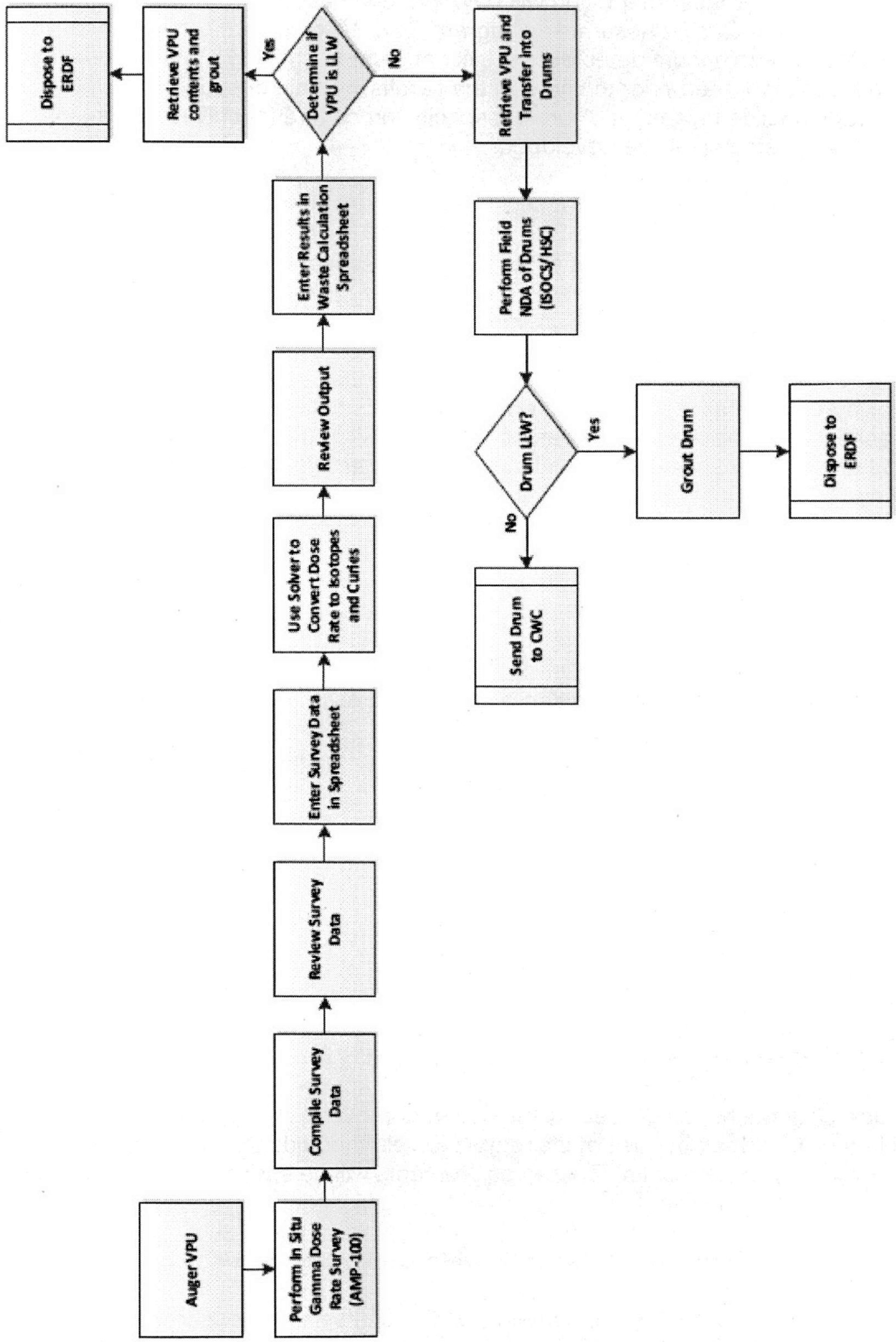
Applicable QA procedures, quantitative target limits, and data quality are dictated by the intended use of the data and characterization methods used. Characterization methods include use of the in situ radiological measurement system to collect dose rates and the field NDA system. Table 3 provides a summary of the analytical methods and equipment used for radiological characterization and/or measurements. Figure 9 is a flowchart of the analytical process for radiological characterization.

**Table 3. Field Analytical Methods for Radiological Characterization.**

| Instrument | Use | Detection Limit | Calibration | QC Requirements | QC Criteria |
|---|---|---|---|---|---|
| AMP-100 (or equivalent) | In situ VPU radiological gamma measurements | 5 mR/hr | Calibration for the energy field to be measured. Site-specific calibration factors will be developed and specified in procedure for use of detector | Source check daily before and after use. Daily background measurement | Within ±20% |
| AMP-50 (or equivalent) | In situ VPU radiological gamma measurements | 0.01 mR/hr | Calibration for the energy field to be measured. Site-specific calibration factors will be developed and specified in procedure for use of detector | Source check daily before and after use. Daily background measurement | Within ±20% |
| ISOCs | NDA of containers (e.g., drums) | Analysis specific | At least annually | Source check daily before and after use. Daily background measurement | Within ±20% |
| HSC | NDA of containers (e.g., drums) | Analysis specific | Daily during use | Source check daily before and after use. Daily background measurement | Within ±20% |

HSC = Hanford slab counter
ISOCs = in situ object counting system

**Figure 9. Analytical Process for Waste Characterization.**



### 3.7.1 Performance Requirements for In Situ Dose Rate Measurements

The in situ radiological instrumentation will be operated in accordance with WCH-314, *Sampling and Characterization Quality Assurance Program Plan, Volume 3: Field Analytical Technical Requirements*. The minimum detectable activity of gamma-emitting radionuclides for the detector will be established prior to use with the results capable of supporting calculations to determine radionuclide inventory. A project-specific procedure for calibration, maintenance, and use of the in situ detector will be developed.

Data quality indicators for use of the in situ radiological instrumentation include the following:

- Precision will be assessed by the collection and assessment of duplicate measurements and performance checks. In general, duplicate field measurements will be collected every 20 measurements. However, this is dependent upon ALARA considerations, the tooling used, and the method of deployment for the instrumentation. For example, the detector will need to be suspended inside the stem of the auger inside the VPU in a manner that approximates the same geometry for both measurements. Duplicate measurements will be addressed in the procedure for operation of the AMP-100 or equivalent detector. The anticipated performance checks and associated objectives, along with collection, frequencies, and corrective actions, will be presented in the procedure for use of the dose rate instrument.

- Accuracy will be addressed through the calibration and maintenance of the dose rate instrument. Calibration and maintenance objectives will be provided in the procedure for use of the dose rate instrument.

  Accuracy for field measurements can be calculated as:

  $$\%C = (A-B/A) \times 100$$

  where

  A = true value
  B = measured value
  C = % percent difference.

- The objective for field measurement completeness is 100%.

- Representativeness is a qualitative parameter that is dependent upon the proper design of the characterization program and is satisfied by ensuring the technical approach is followed and the measurement process conservatively bounds the VPU's radiological inventory.

- Comparability will be maximized by the use of approved WCH procedures; the recording of data in a valid format; the use of standardized field methods; and the reporting of data in appropriate, consistent units. These requirements will be specified in the procedure for use of the dose rate instrument.

### 3.7.2 Performance Requirements for Dose-to-Curie Calculation

Dose-to-curie calculation is used to convert the in situ dose rate measurements to curies of key radioisotopes. The methodology will be finalized in a future calculation.

- The verification and validation report for the MicroShield® computer software is maintained in the 618-10 VPU project file.

- All MicroShield calculations will be peer reviewed by an independent reviewer and signed off. This peer review serves to ensure that the appropriate assumptions are used and to ensure that the calculations are performed correctly.

### 3.7.3 Performance Requirements for Scaling Factors

Scaling factors have been established using historical information and process knowledge. Performance requirements associated with use of scaling factors will be finalized in a future calculation.

### 3.7.4 Performance Requirements for Field Nondestructive Assay

Nondestructive assay equipment is required to perform in a manner that accurately and reliably provides results with sufficient confidence to distinguish TRU waste from low-level radioactive waste. The NDA techniques, instruments, and procedures used must:

- Provide a minimum detectable concentration sufficient to distinguish TRU waste from low-level radioactive waste.

- Provide monitoring for fluctuations in background radiation levels, determining if background levels impact results, and correcting for excessive background radiation, if applicable.
- Account for measurement errors from components such as internal consistency, transmission errors, self-absorption, and/or localized measurement problems.

- Result in defensible values for the activity and mass of the reported radionuclide inventory.

The NDA system must be capable of measuring and reporting results with the following minimum information:

- The measured value, in curies, +/- the uncertainty value calculated at the two-sided 95% confidence level of each isotope of concern detected.

- The minimum detectable activity (MDA) of gamma-emitting isotopes of concern that were not detected by gamma energy analysis.

- Identification and quantification of radionuclides (isotopes of concern) including americium-241, plutonium-239, plutonium-240, uranium-235, uranium-238, and cesium-137 in curies, if detected in the waste. If these radionuclides are not detected, the MDA will be reported.

- Total measurement uncertainty (TMU) for the measured radionuclides.

® MicroShield is a registered trademark of Grove Software, registered in the U.S. and other countries.

Many factors affect the MDA/minimum detectable concentration and TMU reported by an analysis system; for example, the detector to sample calibration geometry, detector resolution, detector efficiency, sample density, sample elemental composition, spatial distribution of activity material, self-attenuation of source materials, containers, energy of the photo peak of interest, and background contributions. The terms lower limit of detection (LLD) and MDA (in units of activity) are used interchangeably. In support of the above requirements, the NDA unit must evaluate, document, and technically justify the following determinations:

1. *Lower Limit of Detection.* The LLD for each NDA system must be determined. Instruments performing TRU waste/low-level waste discrimination measurements must have an LLD of 100 nCi/g or less. Environmental background and container-specific interferences must be factored into LLD determinations. LLD is that level of radioactivity that, if present, yields a measured value greater than the critical level with a 95% probability, where the critical level is defined as that value which measurements of the background will exceed 5% probability. The method(s) for determining LLD will be documented by a qualified analyst.

2. *Quantification of Nondetectable Radionuclides.* Radionuclide quantities that cannot be determined by NDA because there is no method or the MDA is not low enough to support decision making may be scaled to measured radionuclides. This includes the isotopes of concern listed above. Radionuclides expected to be scaled are strontium-90, plutonium-242, and uranium-234. Daughter products below the MDA that are required to be reported will be scaled from the activity of the parent or reported at the MDA (if they can be determined by NDA). The means and methods used to quantify these isotopes from other measured isotopes are to be technically justified. In such cases, the equivalent of an LLD (i.e., reporting threshold for a radionuclide[s], when it is technically justified) will be derived. This value may be based on decay kinetics, scaling factors, or other scientifically based relationships, and must be documented.

3. *Total Measurement Uncertainty.* The method used to calculate the TMU should be documented. Reports may be combined for like or similar systems if the TMU is justified to be identical or if any differences are clearly identified and do not affect the TMU. The likeness or similarity of the systems must be technically justified.

For field NDA, the system will be operated in accordance with WCH-314, *Sampling and Characterization Quality Assurance Program Plan, Volume 3: Field Analytical Technical Requirements*, and RC-300, *Radiological Control Instrumentation Procedures*, Procedure RC-300-4.3, "In-Situ Object Counting System (ISOCS) Quality Assurance."

### 3.7.5 Field Nondestructive Assay Quality Control

Field NDA QC measurements include daily background and source checks and a duplicate measurement per measurement batch.

### 3.7.6 Inspection/Acceptance Requirements for Supplies and Consumables

Procurement activities will comply with current requirements found in BSC-300, *WCH Procurement.* Received items/reagents will be inspected for conformance with specifications defined in the procurement requisition. If the item/reagents do not meet specifications, the item/reagents will be dispositioned through the nonconformance system.

### 3.7.7 Instrument/Equipment Testing, Inspection, and Maintenance Requirements

Field instruments used to take measurements will be tested, inspected, and maintained in accordance with the quality processes and work instructions that satisfy the requirements of the WCH QA program (QA-1, *Quality Assurance*) and WCH-314, *Sampling and Characterization Quality Assurance Program Plan, Volume 3: Field Analytical Technical Requirements.* The processes and work documents will identify the instruments that require testing, inspection, and/or maintenance; specify the frequency; and establish the methods used to test, inspect, and/or maintain each instrument. Correction of nonconformance is performed in accordance with quality processes and work instructions that describe the identification, tracking, and closeout of issues, and satisfy requirements of the WCH QA program (QA-1).

### 3.7.8 Instrument Calibration and Frequency

Onsite instruments used for analysis are calibrated in accordance with WCH-314, *Sampling and Characterization Quality Assurance Program Plan, Volume 1: Administrative Requirements*, and WCH-314, *Sampling and Characterization Quality Assurance Program Plan, Volume 3, Field Analytical Technical Requirements* and as required by WCH-51, *Quality Assurance Program Document*. Results from instrument calibration activities are recorded in a bound logbook in accordance with procedures outlined in ENV-1-2.5, "Field Logbooks," or as specified for radiological measurements. Where applicable, tags will be attached to field screening and onsite analytical instruments to note the date when the instrument was last calibrated and the calibration expiration date.

### 3.7.9 Data Management

Data management includes, but is not limited to, the following:

- The drum (container) identification number and container net and gross weight in kilograms
- A sequenced file number, NDA date, and name and version of any software used for the NDA and data analysis
- The measured value in curies, +/- the uncertainty value calculated at the 95% confidence level of each isotope of concern detected
- The MDA of gamma-emitting isotopes of concern that were not detected by gamma energy analysis
- A report generated for the results of the in situ radiological measurements collected by the AMP-100 or equivalent detector.

Data packages will be reviewed prior to submitting to regulatory agencies or inclusion in reports or technical memoranda, at the direction of the WCH Waste Operations project engineer. Field NDA and in situ radiological measurement data are managed as specified in WCH-314, *Sampling and Characterization Quality Assurance Program Plan Volume 3: Field Analytical Technical Requirements.*

## 3.8 ASSESSMENT/OVERSIGHT

### 3.8.1 Assessments and Response Actions

The WCH Quality Assurance organization may conduct random surveillance and assessments in accordance with QA-1, *Quality Assurance*, Procedure QA-1-1.5, "Self Assessments," to verify compliance with requirements outlined in this SAP, project work packages, WCH procedures, and regulatory requirements.

Deficiencies identified by any of these assessments are reported in accordance with QA-1-1.2, "Corrective Action Management." When appropriate, corrective actions will be taken by the project engineer in accordance with DOE/RL-96-68, *Hanford Analytical Services Quality Assurance Requirements Document*, to minimize recurrence.

### 3.8.2 Reports to Management

It is required that management be made aware of deficiencies identified by assessments or self-assessments. Corrective action required as a result of surveillance reports, nonconformance reports, or audit activities will be documented and dispositioned, as required by QA-1-1.2, "Corrective Action Management." Other measurement systems, procedures, or plan corrections required as a result of routine review processes will be resolved by governing procedures or will be referred to the technical lead for resolution. Findings from audits, surveillance, and assessments will be transmitted to the project manager and the current contractor QA department for program-related tracking and trending. Otherwise, the routine evaluation of data quality described throughout this QAPjP will be documented and filed with the data in the project file.

## 3.9 DATA REVIEW AND VERIFICATION

The final data analysis report for in situ radiological measurements and NDA measurements will be independently reviewed and verified by a qualified individual (see Section 3.3).

## 3.10 FIELD CHARACTERIZATION PROCEDURES

### 3.10.1 In Situ Radiological Measurements

The primary data mechanism for the determination of curie estimates for an augered VPU is radiation surveys or dose rate measurements. Vertically collected in situ dose rate measurements will be collected through the center of the auger flight using an AMP-100 or equivalent detector as specified in an environmental radiological survey task instruction (ERSTI). The ERSTI will be prepared in accordance with procedure ENV-1-2.38, "Preparation of Environmental Radiological Survey Task Instructions." The key dose-emitting radionuclide (cesium-137) will be quantified using the AMP-100 dose rate instrument or equivalent and will serve as a scaling radioisotope. Computer calculation software will be used to transform the dose rate information into the curies present for the scaling radioisotope. Through the use of scaling factors, a source term will be calculated and the quantities of remaining radioisotopes will be determined.

The dose-to-curie calculation is the main process that will be used to translate the field information (dose rate) into a radiological inventory for the key radioisotope, cesium-137. When multiplied by appropriate scaling factors, the radiological inventory for all the radioisotopes (Table 2) can be established. The development of scaling factors and dose-to-curie calculations will be finalized in a calculation brief and will consider factors that affect the calculation results including distance between the radioactivity source and the measurement instrument, background radiation levels, and the effect of shielding on both the source and the instrument. The calculation software will be validated to support use of the in situ radiological measurement procedure.

### 3.10.2 Field Nondestructive Assay Measurements

The augered VPU that is determined to be suspect TRU will be removed and placed in drums. Field NDA of drums will be performed using the procedures for drum characterization specified in WCH-449, *618-10 Burial Ground Drum Sampling and Analysis Instruction*. The NDA measurement techniques are performed in accordance with RC-300, *Radiological Control Instrumentation Procedures,* RC-300-4.3, "In-Situ Object Counting System (ISOCS) Quality Assurance", RC-300-6.6, "Operating Canberra Gamma Spectroscopy Systems", and manufacturer's instructions for the ORTEC and neutron detector instrumentation/equipment. FRC-200, *Field Remediation Closure Technical Procedures*, procedure FRC-200-TP-OOD-001, "Operation of the ORTEC Detective EX," provides instruction for use of the ORTEC, and FRC-200-TP-HSC-001, "Operation of the Hanford Slab Counter," provides requirements for use of the neutron slab counter.

# 4.0 FIELD SAMPLING AND ANALYSIS PLAN

This field sampling and analysis plan provides the characterization methods that will be implemented to supplement the in-process information and technical evaluations, as discussed in Section 1.4.

## 4.1 CHARACTERIZATION PROCESS DESIGN

### 4.1.1 Identification of Vertical Pipe Units for Characterization

As part of the DQO process, the following methodology was agreed upon for determining the VPU TRU concentration and waste classification:

- Rank the VPUs using an analysis of the NIC data and the inventories developed for the 618-10 Burial Ground.

- Based on the ranking, select two consecutively ranked VPUs to perform in situ radiological characterization.

- Evaluate the results and determine if each VPU meets the ERDF WAC (pass) or does not (fail). The possibilities for the two consecutively ranked VPUs are pass/pass, pass/fail, fail/pass, or fail/fail.

- At the ranking point where two consecutively ranked VPUs pass, then subsequent lower ranked VPUs are thereby determined to all pass and do not require in situ radiological characterization.

- If at least one of the two consecutively ranked VPUs fails, then VPUs above that ranking point would be considered to fail unless specifically characterized.

- A minimum of four VPUs will be characterized.

If the augered VPU is determined to meet the low-level radioactive waste criteria, it will be grouted and then disposed at the ERDF. A VPU not meeting the ERDF WAC will be retrieved in drums for further evaluation. Drums that are determined to meet the low-level radioactive waste criteria will be grouted and then disposed at the ERDF; otherwise, the drums will be processed as suspect TRU for storage/disposal at the CWC as required by HNF-EP-0063, *Hanford Site Solid Waste Acceptance Criteria.*

#### 4.1.2 Vertical Pipe Unit Characterization Methodology

See Section 3.10.1.

#### 4.1.3 Suspect Transuranic Characterization Methodology

Field NDA is discussed in Section 3.10.2. Drums determined by field NDA to be low-level waste will be prepared for shipment to ERDF. Drums determined to be suspect TRU will be submitted to the CWC and must meet the requirements for acceptance identified in HNF-EP-0063. These include meeting the requirements for package dimension, weight, and dose rate.

### 4.2 FIELD DOCUMENTATION AND SAMPLE MANAGEMENT

#### 4.2.1 Field Documentation

Field documentation is kept in accordance with the following procedures:

- Procedure ENV-1-2.5, "Field Logbooks"

- RC-300, *Radiological Control Instrumentation Procedures*, procedure RC-300-4.3, "In-Situ Object Counting System (ISOCS) Quality Assurance."

- Procedure for use of the AMP-100 or equivalent detector.

#### 4.2.2 Suspect TRU Transport

All suspect TRU drums will be packaged and shipped in accordance with U.S. Department of Transportation and/or DOE/RL-2001-36, *Hanford Sitewide Transportation Safety Document*, requirements.

## 4.3 QUALITY CONTROL REQUIREMENTS

Quality control procedures must be followed in the field to ensure that reliable data are obtained. When performing this field characterization effort, care is taken to prevent the cross-contamination of equipment that could compromise data integrity.

### 4.3.1 Field Nondestructive Assay Quality Control

Quality control requirements for field NDA are specified in the operational procedures. Field NDA QC requirements are consistent with requirements specified in PRC-RD-EN-10484, *Nondestructive Assay Management Program* (WCH 2012).

### 4.3.2 In Situ Field Dose Rate Instrumentation Quality Control

Quality control requirements for in situ instrumentation will be specified in the procedure for operation of the AMP-100 or equivalent detector.

## 4.4 INSTRUMENT CALIBRATION AND MAINTENANCE

Instrument calibration and maintenance is conducted in accordance with the QC requirements identified in each measurement method standard operating procedure and QA plan, and the manufacturer's instructions.

### 4.4.1 In Situ Radiological Measurements

All calibration procedures and measurements will be made in accordance with manufacturers' specifications, contractor standard operating procedures, and WCH-314, *Sampling and Characterization Quality Assurance Program Plan, Volume 3: Field Analytical Technical Requirements*. Field instruments will be checked and calibrated before their use on site, and batteries will be charged and checked daily where applicable. Instrument response checks for the AMP-100 will be performed against a sealed source of known activity at the beginning and end of each workday. If deemed necessary by characterization personnel or if response checks identify a discrepancy in performance, the instrument will be taken out of service and submitted for calibration according to manufacturers' specifications. Equipment that fails calibration and/or becomes otherwise inoperable during the field investigation will be removed and either repaired or replaced.

All documentation pertinent to the calibration and/or maintenance of field measurement equipment will be recorded in a field logbook. Logbook entries regarding the status of field equipment will contain, but will not necessarily be limited to, the following information:

- Date and time of calibration
- Name of person conducting calibration
- Type of equipment being calibrated (make and model)
- Reference standard used for calibration.

### 4.4.2 Nondestructive Assay Measurements

Instrument calibration and maintenance requirements for measurements are specified in WCH-314, *Sampling and Characterization Quality Assurance Program Plan, Volume 3: Field Analytical Technical Requirements*; RC-300, *Radiological Instrumentation Procedures*, procedures RC-300-4.3, "Quality Management Program for the In-Situ Object Counting System (ISOCS)" and RC-300-6.6, "Operating Canberra Gamma Spectroscopy Systems"; and manufacturer's instructions for the ORTEC and neutron detector instrumentation/equipment. FRC-200, *Field Remediation Closure Technical Procedures*, procedure FRC-200-TP-OOD-001, "Operation of the ORTEC Detective EX," provides instruction for use of the ORTEC, and procedure FRC-200-TP-HSC-001, "Operation of the Hanford Slab Counter," provides requirements for use of the neutron slab counter.

# 5.0 HEALTH AND SAFETY

All field operations will be performed in accordance with WCH health and safety requirements, which are outlined in SH-1, *Safety and Health*, and RC-1, *Radiation Protection Procedures.*

Work planning, hazards analysis, and contingency planning will be conducted in accordance with the work control process as described in PAS-2, *Integrated Work Control Program.* The project work packages will include a job hazard analysis, site-specific health and safety plan, and applicable radiological work permits.

The in situ measurement procedures and associated activities will consider exposure reduction and contamination control techniques that will minimize the radiation exposure to the characterization team as required by RC-1, QA-1, and SH-1.

# 6.0 MANAGEMENT OF WASTE

Waste generated by characterization activities will be managed in accordance with WMT-1, *Waste Management and Transportation.*

# 7.0 REFERENCES

0600X-CA-N0083, 2012, *Radiological Inventory in the 618-10 Vertical Pipe Units*, Rev. 0, Washington Closure Hanford, Richland, Washington.

0600X-CA-N0098, 2013, *618-10 Vertical Pipe Units Ranked by TRU Concentration*, Rev. 2, Washington Closure Hanford, Richland, Washington.

0600X-CA-N0100, 2012, *618-10 and 618-11 Burial Ground Radiological Source Terms*, Rev. 0, Washington Closure Hanford, Richland, Washington.

10 CFR 61.55, "Waste Classification," *Code of Federal Regulations*, as amended.

10 CFR 830, "Nuclear Safety Management, Subpart A, Quality Assurance Requirements," *Code of Federal Regulations,* as amended.

ASTM C1490, *Standard Guide for Selection, Training, and Qualification of Nondestructive Assay (NDA) Personnel*, ASTM International, West Conshohocken, Pennsylvania.

BHI-00291, 1997, *Geophysical Investigations of the 618-10 and 618-11 Burial Grounds, 300-FF-2 Operable Unit*, Rev. 1, Bechtel Hanford, Inc., Richland, Washington.

BSC-1, *Business Services and Communications*, Washington Closure Hanford, Richland, Washington.

BSC-300, *WCH Procurement*, Washington Closure Hanford, Richland, Washington.

DOE O 414.1C, *Quality Assurance*, as amended, U.S. Department of Energy, Washington, D.C.

DOE/RL-96-68, 2007, *Hanford Analytical Services Quality Assurance Requirements Documents*, Rev. 0, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

DOE/RL-2001-36, 2011, *Hanford Sitewide Transportation Safety Document*, Rev. 1-E, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

DOE/RL-2001-48, 2014, *300 Area Remedial Action Sampling and Analysis Plan*, Rev. 4, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

DOE/RL-2008-27, 2009, *Sampling and Analysis Plan for 618-10 and 618-11 Nonintrusive Sampling*, Rev. 0, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

ENV-1, *Environmental Monitoring & Management*, Washington Closure Hanford, Richland, Washington.

EPA/240/B-01/003, 2001, *EPA Requirements for Quality Assurance Project Plans,* EPA QA/R-5, U.S. Environmental Protection Agency, Office of Environmental Information, Washington, D.C.

EPA/240/B-06/001, 2006, *Guidance on Systematic Planning Using the Data Quality Objectives Process*, EPA QA/G-4, U.S. Environmental Protection Agency, Office of Environmental Information, Washington, D.C.

EPA/240/R-02/009, 2002, *Guidance for Quality Assurance Project Plans*, EPA QA/G-5, U.S. Environmental Protection Agency, Office of Environmental Information, Washington, D.C.

FRC-200, *Field Remediation Closure Technical Procedures,* Washington Closure Hanford, Richland, Washington.

HNF-EP-0063, 2010, *Hanford Site Solid Waste Acceptance Criteria*, Rev. 15 (or latest revision), CH2MHill Plateau Remediation Company, Richland, Washington.

North Wind, 2010, Hanford-RCCC Subcontract C008013A00, *Non-Intrusive Characterization of 618-10 Burial Ground Submittal 5-02, Data Summary Report*, NWI-WA-2010-84, North Wind, Idaho Falls, Idaho.

PAS-2, *Integrated Work Control Program*, Washington Closure Hanford, Richland, Washington.

PRC-RD-EN-10484, "Nondestructive Assay Management Program," CH2MHill Plateau Remediation Company, Richland, Washington.

QA-1, *Quality Assurance*, Washington Closure Hanford, Richland, Washington.

RC-1, *Radiation Protection Procedures*, Washington Closure Hanford, Richland, Washington.

RC-300, *Radiological Control Instrumentation Procedures*, Washington Closure Hanford, Richland, Washington.

SH-1, *Safety and Health*, Washington Closure Hanford, Richland, Washington.

WCH, 2009a, *618-10 Burial Ground Geophysical Summary Focusing on Metal Concentrations*, Geophysical Investigation 0602202, Washington Closure Hanford, Richland, Washington.

WCH, 2009b, *Integrated Chemical and Radiological Hazard Evaluation Worksheet – 618-10 Burial Ground/North Burial Ground,* IHE-2009-0006, Rev. 3, Washington Closure Hanford, Richland, Washington.

WCH, 2012, "Evaluation of Washington Closure Hanford's General Procedural Requirements and Non Destructive Assay Program Compared to the CH2M HILL Plateau Remediation Company Non Destructive Assay Management Program," CCN 163476 to File from E. S. Wallace, Washington Closure Hanford, Richland, Washington, January 3.

WCH, 2014, "618-10/11 Regulatory Interface Meeting, February, 2014," CCN 174664 to Distribution from D. Faulk, Washington Closure Hanford, Richland, Washington, February 24.

WCH-51, 2013, *Quality Assurance Program Document*, Rev. 9, Washington Closure Hanford, Richland, Washington.

WCH-191, 2014, *Environmental Restoration Disposal Facility Waste Acceptance Criteria*, Rev. 3, Washington Closure Hanford, Richland, Washington.

WCH-314, 2009, *Sampling and Characterization Quality Assurance Program Plan, Volume 1: Administrative Requirements*, Rev. 0, Washington Closure Hanford, Richland, Washington.

WCH-314, 2009, *Sampling and Characterization Quality Assurance Program Plan, Volume 3: Field Analytical Technical Requirements*, Rev. 0, Washington Closure Hanford, Richland, Washington.

WCH-431, 2011, *Field Investigation Report for the 618-10 Burial Ground Intrusive Sampling*, Rev. 0, Washington Closure Hanford, Richland, Washington.

WCH-449, 2013, *618-10 Burial Ground Drum Sampling and Analysis Instruction*, Rev. 4, Washington Closure Hanford, Richland, Washington.

WCH-459, 2015, *Documented Safety Analysis for Remediation of the 618-10 Burial Ground*, Rev. 2, Washington Closure Hanford, Richland, Washington.

WCH-525, 2012, *Vertical Pipe Unit Disposition for 618-10 and 618-11 Burial Grounds*, Rev. 0, Washington Closure Hanford, Richland, Washington.

WCH-550, 2015, *618-10 Burial Ground VPU Treatment* Plan, Rev. 0, Washington Closure Hanford, Richland, Washington.

WMT-1, *Waste Management and Transportation*, Washington Closure Hanford, Richland, Washington.

# DISTRIBUTION

U.S. Department of Energy
<u>Richland Operations Office</u>

| | |
|---|---|
| J. Zeisloft (3) | A3-04 |

U.S. Environmental Protection Agency
<u>Hanford Project Office</u>

| | |
|---|---|
| B. Simes | B1-46 |

<u>Washington Closure Hanford</u>

| | |
|---|---|
| S. W. Bisping | T2-05 |
| W. A. Borlaug | T2-03 |
| M. A. Buckmaster | N2-05 |
| H. M. Carman | N2-05 |
| S. W. Callison | N2-05 |
| R. Coffman | N2-02 |
| R. A. Carter | N3-30 |
| M. A. Casbon | T2-03 |
| R. J. Garrett | H4-20 |
| B. M. Gillespie | N2-02 |
| D. L. Haggard | N2-02 |
| R. T. Hynes | N3-20 |
| K. J. Koegler | N3-30 |
| M. E. Lewis | T2-05 |
| J. D. Ludowise | N2-02 |
| D. B. Teachout | N3-30 |
| W. S. Thompson | H4-23 |
| R. L. Weiss | H4-21 |
| J. L. Westcott | N2-02 |
| S. G. Wilkinson | N3-30 |
| A. E. Zacharias | N2-05 |

| | |
|---|---|
| Document Control | H4-11 |
| DOE-RL Public Reading Room | H2-53 |
| Hanford Technical Library | K3-78 |